

# Argonne National Laboratory

A PACKAGED LOOP FOR EBR-II:  
PRELIMINARY CONCEPTUAL-DESIGN STUDIES  
AND DISCUSSION OF POSSIBLE FUTURE WORK

by

Charles W. Wilkes, Ralph E. Rice, Jr.,  
and Harold E. Adkins

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EBR-II Project

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ABSTRACT

A conceptual design of a packaged loop, which would be placed in one of the EBR-II control-rod positions, has been developed. The loop would have its own closed-circuit coolant system, which would be isolated from the reactor coolant. The capability of such a loop for higher temperatures, isolation from the reactor coolant, containment of failures, and independent control of power density, coolant flow, and sodium chemistry would permit the performance of several types of experiments that could not normally be conducted in the reactor.

I. INTRODUCTION

This report documents the conceptual-design studies that were conducted for a packaged loop for EBR-II. Not all of the conceptual-design studies were completed, nor was a conceptual system design description prepared. However, a preliminary feasibility and cost study<sup>1</sup> was published in October 1968. Effort on the project was suspended in March 1969 because of funding limitations and the concern of Experimental Breeder Reactor II (EBR-II) Project management that:

1. The crowding of equipment on the small rotating plug of the reactor would increase the difficulty of maintaining the control-rod drives, the fuel-handling mechanisms, the kinetics-testing rod drive, and the instrumented subassembly. These mechanisms, as well as the packaged loop, the in-core instrument test facility, and their shielded handling machines, would all have their centerlines on a circle about 18 in. in diameter. The increased difficulty of access for maintenance would have a potential for increasing reactor downtime.

2. The time required for component development, design, fabrication, and prototype testing to achieve a reliable packaged loop could detract from its programmatic desirability because of late availability.

3. The cost of a complete packaged-loop project might cause funding problems in the near future when serious efforts are being made to economize.

## II. SUMMARY

### A. Justification

The Liquid Metal Fast Breeder Reactor (LMFBR) programs for core design, fuels and materials, and sodium technology contain plans for experiments that could be best performed in an EBR-II packaged loop. Because of its planned capabilities for higher temperature, isolation from the reactor system, failure containment, and independent control of power density, coolant flow, and coolant chemistry, the loop would permit the performance of several types of experiments that normally could not be conducted in the reactor. Experiments that could be conducted in the loop are listed below in the order of their present priority in the LMFBR Program Plans:

1. "Off-nominal" testing of fuels and materials at higher power densities, temperatures, burnups, coolant-impurity levels, etc., to establish practical limits.
2. Steady-state irradiation of fuel elements and materials under future LMFBR conditions of temperature, power density, and coolant chemistry to point of failure to determine operating limits.
3. Continued steady-state irradiation of failed or defected fuel elements to determine consequences of failure.
4. Sodium-technology experiments.
5. Vented-fuel experiments.

### B. Safety Considerations

If further funding is provided, the present feasibility studies on the EBR-II packaged loop will be followed by a detailed engineering study of an actual design. The engineering study would include various trade-off studies encompassing fabrication, installation, operation, and safety problem areas of the packaged loop. The detailed design would be expected to conform to the safety considerations listed in Section IV. Additional safety-design criteria would be formulated as necessary to ensure a high degree of inherent safety in the packaged loop during operation in the EBR-II core. The proposed research and development program would aid in assessing various operational problem areas that involve safety unknowns. The completed packaged loop for EBR-II would be a balanced trade-off between fabrication and installation difficulties, operational variables, and overall plant-safety requirements.

### C. Description of Facility

According to the initial plans, the in-pile experimental loop assembly would be inserted into a control-rod thimble through the small rotating plug and the reactor-vessel cover. This assembly would be elevated above the reactor core during fuel handling to prevent interference with the fuel-handling operation. Two concepts of the loop have been considered: an internal-pump concept and an external-pump concept. The more promising of these, the internal concept, locates the loop pump within the small-diameter in-pile tube, allowing the main sodium flow to stay within the primary-tank shielding. Only small quantities of sodium required for sampling and purification would be routed outside the primary-tank shielding.

The development of an internal pump would require the expenditure of some effort and time. The cost in time and money will be assessed before proceeding with development. A prototype 1000°F annular linear-induction pump is currently being developed for sodium sampling service. Experience with this pump would serve as a basis for the design of the packaged loop pump.

The other concept locates the loop pump along with a considerable amount of sodium above the primary-tank shielding.

With either concept, a special shielded handling machine would be required for installing and removing the in-pile tube and its contents. Specific design criteria for the loop are summarized in Table I.

TABLE I. Design Criteria for EBR-II Packaged Loop

Coolant	Sodium
Design temperature	
Loop and components	1200°F
Test section	1475°F
Design pressure	
Test section	1375 psi
Above test section	475 psi
Maximum power-rejection capability	>535 kW
Flow (variable)	10-70 gpm
Maximum pump requirements	70 gpm at 100 psi
Experimental capacity--maximum OD of test cross section	1.7 in.

#### D. Effect on Operations

The installation, maintenance, normal operation, removal, and reinstallation of the packaged loop would have some effect on the EBR-II plant factor. Any engineering work undertaken in the future would proceed with an objective of establishing a conceptual design that would have a minimum effect on the plant factor.

#### E. Research and Development (Critical Problem Areas)

The area of most concern with the EBR-II packaged-loop concept is the removal of a completed experiment, which might entail distribution of plutonium-bearing fuel throughout the cooling system of the loop. Because of limited shielding space above the control-rod drive nozzle, operators would have to (a) use distance and personnel shielding for protection during a part of the removal operation, or (b) use a considerable amount of shielding between the rotating plug and the shielded handling machine. The reliability of handling equipment would have to be established to minimize the possibility of equipment failures during removal operations.

In the present conceptual design, a potential interference may exist between the shielded handling machine (when it is in position to receive an irradiated experiment) and the instrumented subassembly elevating mechanism. This problem would have to be resolved during detailed engineering design to eliminate any interference with the instrumented subassembly.

Other areas that require research and development are discussed in Section VI. These areas are (a) internal pumps, (b) seals and connections, (c) purification and sampling equipment, (d) heat exchangers, (e) handling equipment, and (f) shielding requirements.

### III. JUSTIFICATION

#### A. LMFBR Program Plans for EBR-II Packaged Loop

The LMFBR programs for core design, fuels and materials, and sodium technology contain plans for experiments that could be best performed in an EBR-II packaged loop. In fact, the number of EBR-II loop-type experiments mentioned in the LMFBR Program Plans exceeds the capacity of the two possible loops, and experimental priorities would have to be established. This section of the study contains a summary survey of program plans for the possible EBR-II loops and of the experiments that might be conducted in the loops. Statements pertaining to EBR-II packaged loops are underlined. Additional comments are included as noted in brackets [ ].

##### 1. Core Design Program

From the Core Design Program Plan (Vol. 6),<sup>2</sup> the following is quoted relative to the possible use of EBR-II packaged loops:

"Sub-Task 6-4.2.5 Development of Packaged Loops: The major purpose of this sub-task is the development and manufacture of packaged loops for use in EBR-II. The work will include a value analysis to compare the worth of the technical information obtained with the risks and losses in plant factor incurred by the use of packaged loops in EBR-II.

"This sub-task will start with the current concept as defined in the EBR-II Five-Year Plan and proceed through the various development stages to produce a proven design. The packaged loop will be inserted in a control rod thimble. This sub-task includes the fabrication and maintenance of a stock of components for the operational end item to satisfy the needs of experimenters.

"The following actions will be performed to implement this sub-task.

"(1) Complete the present conceptual design, including all engineering, safety, and reliability analysis.

"(2) Detail the design and manufacture and test (out of the reactor) of one or more development units.

"(3) Refine the design and build and install in the reactor a prototype unit for a limited period of operation.

"(4) Produce components to permit assembly of units for use by experimenters.

"(5) Develop and manufacture the auxiliary handling equipment for removal of the experimental units.

"This sub-task has a major interface with the Fuels and Materials element as well as with Sub-task 6-4.3.2. [FFTF Packaged Loops]

"The schedule calls for completion of the prototype unit in FY 1971. This will be followed by installation of units that will continue to be used until FFTF becomes operational and similar facilities are made available about 1974.

"Implementation currently is lagging and if it is not accelerated, there will be little beneficial experimental use before FFTF becomes available. However, the development will still be an excellent pilot project for the FFTF loops to follow.

"Priority: 2; FY 69/78."

From Appendix 3--Requirements and Capabilities for Irradiation Testing of Ref. 2, the following is quoted as being the type of tests that are programmed for the EBR-II packaged loops:

"A significant portion of the technology needed by core designers must be obtained in a neutron radiation environment. The successful performance of a reactor core component under all the conditions that it is expected to survive, is the ultimate test of an adequate design. Fast reactor technology development requires fast flux irradiation facilities with the capability of testing under steady-state, off-nominal, to failure, and beyond failure conditions. [Note: Off-nominal operation increases the probability of failure, and the other two conditions require operation to failure. The performance of many of these types of tests in EBR-II requires a packaged loop.]

"The irradiation testing requirements for LMFBRs are stipulated in the technical programs of the Core Design, Fuels and Materials, Sodium Technology, and Safety elements. These requirements can be categorized as follows:

"(1) Steady-State Testing. This involves primarily the accumulation of a prescribed level of fuel burnup (or irradiation exposure) at prescribed conditions of [high] temperature and [high] burnup (or exposure) rate. It also includes startup, normal and emergency shutdown, and occasional departures from nominal power conditions that are indigenous to the test reactor and may or may not be relevant to the ultimate application.

"(2) Off-Nominal Testing. This involves the deliberate operation at power levels, temperatures, transient rates, number of cycles, etc., that exceed those in steady-state testing. Such testing may be done to confirm the ability of a core component to survive such conditions or to seek the survivable limit of such conditions.

"(3) Tests-to-Failure. This involves the accumulation of burnup (or irradiation exposure), generally under steady-state operating conditions, not to a prescribed level but rather to the point of failure of the core component.

"(4) Failure-Effects Testing. This involves testing past the point of failure or testing deliberately defected components for purposes of establishing the consequences of continued operation with failed core components. Post-failure operation may include both steady-state and off-nominal conditions.

[NOTE: These types of tests can be accomplished in an EBR-II packaged loop.]

"A package loop concept has been developed but is not as advanced as the instrumented assembly. This device will permit instrumented tests in a controlled-coolant environment of multiple-element assemblies (number depending on diameter and P/D ratio). Again, the control rod positions will be needed.

#### "Off-Nominal Testing

"Other off-nominal testing (besides transients) is also important. The steady-state fast reactor facilities--EBR-II and Fermi--could be used for such tests as (1) overpower operation for extended periods; and (2) reduced flow operations for extended periods. Unfortunately, these two (and almost any other such off-nominal test) will introduce unacceptable (at present) probabilities of fuel element rupture. Therefore, such experiments must probably be reserved for special loop facilities in the EBR-II or thermal reactors.

#### "Tests-to-Failure

"Tests-to-failure are defined as tests under the usual steady-state irradiation conditions where an appreciable fraction of failures is anticipated. In some cases such experiments may run until all of the specimens have failed. The usual test variable would be burnup; however, specific power, coolant temperature, and other variables known or suspected to affect failure rates could also be chosen. All such tests would require an appreciable time. The criteria for failure could vary but would normally be loss of hermeticity of the cladding. The rationale for testing to failure as an efficient tool for assessing the reliability of a fuel-element design is explained in Appendix 2.

"EBR-II, as presently constructed and operated, cannot be used for deliberate tests-to-failure, nor can it be used for testing of vented fuel

elements. However, the proposed packaged loop will provide a limited capacity for tests of these types.

"Full and complete instrumentation is required for tests-to-failure. Particularly important are the detection of the instant of failure and the measurement of extent of failure. Instrumentation for these purposes is in an early state of development and of uncertain status as far as reactors in general are concerned. The problem is simpler for single fuel elements or experimental assemblies where the mode of failure is known.

### "Failure-Effects-Testing

"All fuel element failures must be considered to have some effect on adjacent components until proven otherwise. A considerable amount of information on these effects will result from the type of testing described under the preceding headings. In those, however, the primary goal is to determine performance limits of fuel elements. In the case of failure-effects testing, the goal is to determine the effects of failure on continued performance of both the affected component and adjacent components.

"Failure effects, at present, are poorly understood even for fuel-element types that have received the greatest attention. Since some failures are inevitable in the course of operating tens of thousands of fuel elements to economically reasonable levels of burnup, it is important to know, for example, which of the following statements is most nearly true:

"(1) An initial random failure will propagate instantly and wide-spread damage will occur before the scram signal can be given.

"(2) Failures will not propagate instantly, but if normal operation is continued, the cladding breach will gradually open, fuel will wash out, and secondary failures will occur by channel blockage.

"(3) Failures are passive both at the instant of their occurrence and indefinitely thereafter, so there is no need to locate and remove the assembly involved prior to its prescheduled discharge burnup.

"Facilities for this vital area of testing are almost nonexistent. As discussed previously, EBR-II and the Fermi reactor are not equipped to allow failure." [Note: A packaged loop in EBR-II could accomplish failure-effects testing on limited-size fuel bundles.]

## 2. Fuels and Materials Program

From the Fuels and Materials Program Plan (Vol. 7),<sup>3</sup> the following excerpts are quoted regarding the use of EBR-II Packaged Sodium Loops in the LMFBR fuels development program:

## "Plan of Action

"The Fuels and Materials element of the Program Plan is divided into three major areas of development: (1) near-term fuel-element materials; (2) long-term fuel-element materials; and (3) other materials (blankets, absorbers, moderators, and structural and special purpose materials). The efforts for near- and long-term fuel elements, respectively, entail development of:

"(1) A highly reliable mixed-oxide stainless steel-clad fuel element for FFTF and early demonstration LMFBRs.

"(2) High-performance fuel elements for the commercial LMFBR plants. Other fuel materials, in addition to mixed oxides, will be considered for these plants. Such fuels may be used in second-generation cores of demonstration LMFBRs, if they qualify for this application by meeting the goals established for commercial-plant fuels.

### "Fuel Element Behavior and Proof of Design Concepts for Near-Term Missions (Tasks 7-3.1 and 7-3.2)

"To prove near-term design concepts and to detect unfavorable fuel-element behavior, assemblies containing simulated fuel elements as near to prototype as test facilities will permit will be tested under expected operating conditions.

### "Behavior of Defected Fuel Elements and Accentuated Failure Testing for Near-Term Missions (Tasks 7-3.3 and 7-3.4)

"There is insufficient information on the failure threshold of mixed-oxide-stainless steel-clad fuel elements. Information is also needed concerning the continued operation of the defective fuel elements that must be expected in any reactor. For the mixed-oxide, stainless steel-clad fuel elements for near-term reactors, it must be shown that, when coolant sodium enters through a defect: (1) excessive swelling of the fuel does not take place; and (2) excessive fuel does not escape from the fuel element.

"Fuel elements will be tested to failure under accelerated conditions to determine the mechanisms of failure. Intentionally defected fuel elements will be tested to demonstrate whether or not they can continue to operate satisfactorily, within prescribed reactor operating limits, after sodium has entered through a defect or failure.

### "Fuel Swelling and Gas Release (Task 7-2.1)

"Sub-Task 7-2.1.1 Intrinsic Swelling Rate: The intrinsic swelling rate will be determined as a function of temperature, flux, specific power, burnup and degree of restraint.

"The effects on swelling of various relations between specific power and thermal and fast fluxes will be studied between 500°C (930°F) and fuel melting. Degree of restraint will be varied between strong cladding (gun-barrel experiments) and thin cladding (almost unrestrained).

"Under this Priority 1 sub-task, work is planned from FY 1969-FY 1974.

"Sub-Task 7-2.1.2 Void Deployment: The effect of void deployment on the swelling behavior will be determined at high burnups both in thermal and fast reactor environments. To be evaluated are porous solid pellets, annular gaps around high-density solid pellets, dished pellets, and central holes in high-density pellets. The swelling behavior will be correlated with fuel-operating temperatures--500°C (930°F) to melting. Fission-gas release will be measured as an integral part of the experiments. The relationship between gas release, swelling behavior, and burnup will be determined. These tests will be standardized as much as possible to provide the maximum degree of cross-comparison of fuel materials among different experimenters. The information obtained will form the basis for selection of materials and deployment methods for integral tests in a fast flux environment (see Task 7-3.1 through 7-3.5).

"Under this Priority 1 sub-task, work is planned from FY 1969-FY 1974.

[Note: The EBR-II packaged loops can be used to achieve the range of fuel temperatures mentioned here.]

#### "Task 7-2.6 Fuel-Sodium Compatibility"

"In near-term fuel-element designs, cladding failure is likely to result in sodium contacting the fuel. To assess possible unacceptable fuel behavior following cladding failures, the consequences of such contact must be known.

##### "Objective"

"The objective of this task is determination of the compatibility between mixed-oxide fuel and sodium under LMFBR conditions.

##### "Scope"

"This task is concerned with the behavior of mixed-oxide fuel in helium-bonded fuel elements that has come into contact with sodium because of cladding failure. Problems associated with sodium-bonded fuel elements are of long-range interest only and are covered in Task Area 7-6.

"The basic behavior of mixed oxide in sodium is covered in Sub-Task 5-6.3.7 of the Sodium Technology element of the Program Plan. The transfer of fuel constituents or fission products into the coolant sodium from defected fuel elements is of direct concern to Task 5-3.1 of that element.

"This task is concerned with both out-of-reactor and in-reactor experimental work. The variables that will be investigated for their effect on fuel-sodium compatibility are fuel composition (O/M ratio and plutonium content), fuel density, temperature, fuel burnup, and sodium impurity level (particularly oxygen).

"Sub-Task 7-2.6.2 In-Reactor: Specimens irradiated in defected-element tests under Task 7-3.3 will be examined to provide information on the effects of fission products, flux, fluence, and thermal gradients on compatibility to supplement and verify the out-of-reactor studies of Sub-Task 7-2.6.1.

"Under this Priority 1 sub-task, work is planned for FY 1969-FY 1974.

### "Task 7-3.3 Defected-Fuel-Element Behavior

"Defective fuel elements can be expected in any reactor, no matter how elaborate the preventive measures. The ability of the reactor to continue normal operation depends on the behavior of defective fuel elements and the fuel in contact with sodium.

#### "Objective

"The objective of this task is an understanding of the behavior of defective fuel elements under continued reactor operation.

#### "Scope

"This task is limited to the behavior of defected or failed fuel elements under normal and slightly off-normal conditions. Both instantaneous effects and the effects of continued operation are of interest.

"These studies will be coordinated with the investigations of sound fuel elements under normal and off-normal conditions (Task 7-3.2) and under accentuated conditions leading to failure (Task 7-3.4)... . The Fuels and Materials element is concerned with gross loss of fuel from the fuel element, while the Sodium Technology element is interested in the precise information on rates at which fission products enter the sodium coolant. To avoid duplication of experiments in the limited loop facilities that are available, coordination of the efforts of both sections is required... .

### "Plan of Action

"This task is concerned with demonstrating whether a defective fuel element can remain in the core, without propagating failure, until the next scheduled reactor shutdown. Of interest, therefore, are sodium logging, fuel loss, conductivity changes, compositional changes and dimensional changes under steady-state and transient conditions. Fuel-element designs as close as possible to those for prototype FFTF and demonstration LMFBRs will be used.

"Sub-task 7-3.3.3 Effects of Continued Operation: The effects of continued operation on the behavior of defected fuel elements and on the behavior of immediately adjacent fuel elements will be determined. Tests in this sub-task will be continued only if corresponding tests under transient conditions (Sub-task 7-3.3.2) do not result in propagation of failures.

"The risk level of the initial tests precludes the use of EBR-II.

[Note: This type of test could be safely run in an EBR-II packaged loop.]

### "Task 7-3.4 Tests to Accentuate Failure

#### "Objective

"The objective of this task is the establishment of the failure threshold of fuel elements.

#### "Scope

"This task provides for the testing of sound fuel elements under accentuated conditions until failure occurs. This task will characterize the conditions under which failure of irradiated and unirradiated oxide-fuel elements can be expected and, if possible, will determine the mechanism of failure. The scope includes confirmation of the design margin and of the ability of fuel elements to survive credible malfunctions and off-normal conditions without undergoing effects that limit their useful life.

### "Plan of Action

"Tests of sound fuel elements under accelerated failure conditions will be conducted both in and out of reactor. The methods used to develop an understanding of failure mechanisms through accelerated tests will be developed by the experimenter.

"The tests will be designed to:

"(1) Define the limiting conditions of fuel and cladding that tend to end the useful life of prototypical fuel elements (by cladding failure, gross fuel redistribution, etc.)

"(2) Isolate the mode of failure.

"The estimated cost for work in this task is between \$2.4 and \$3.6 million.

"Sub-task 7-3.4.3 Fast Flux: The effects of fuel-element geometry and fuel parameters--in the range given in Table 7-III--on fuel-element life under accelerated testing conditions in a fast flux will be determined. The tests will be performed in temperature-controlled, sodium-filled capsules in EBR-II. Later use of instrumented assemblies is possible. Closed-loop facilities planned for the FFTF will be used for more extensive testing. Transient tests will be conducted in TREAT in the near future.

"Under this Priority 1 sub-task, work is planned for FY 1969-FY 1977.

[Note: An EBR-II packaged loop could also be used for this experiment.]

#### "Task 7-4.2 Mark II

"The limited burnup capability of the Mark I series of fuel elements has caused undesirably high consumption of fuel assemblies. This consumption, in turn, has decreased the plant operating factor by requiring short runs and more frequent shutdown for refueling. Clearly, a more advanced fuel-element design is required.

#### "Objective

"The objective of this task is a reliable EBR-II driver fuel element that: (1) permits 2 a/o burnup or greater; (2) can safely operate at 62.5 MWt limited only by the fuel-cladding interface reaction temperature; (3) has expansion volumes sized for full equilibrium swelling; and (4) in its burnup potential is not subject to uncertainties and variations in the swelling rate.

#### "Scope

"This task will be confined to the design and testing, under normal and off-normal conditions, of a fuel element--using Types 304L or 316 stainless steel-clad U-5 Fs alloy--with a large sodium annulus to accommodate fuel swelling and a large gas reservoir designed for full gas release at high burnups.

### "Plan of Action

"The reference Mark II element will be irradiation-tested to determine if it meets the objectives of this task. Subsequently, statistical assurance of the performance requirements will be obtained by a statistical proof-testing program. The behavior of fuel elements under off-normal conditions and the consequences of operation with defective fuel elements will be investigated to the extent needed to determine whether serious problems can be expected.

"Sub-Task 7-4.2.4 Defected Fuel Element Behavior: Work will proceed as under Sub-Task 7-4.1.3.

"Under this Priority 1 sub-task, work is planned for FY 1969-FY 1971.

### "Task 7-4.3 Improved Mark II

#### "Plan of Action

"An improved Mark II design will be developed and tested to establish whether it meets the objectives of this task. Subsequently, the performance requirements will be assured by statistical proof testing.

"The behavior of fuel elements under off-normal conditions and the consequences of operation with defective fuel elements will be investigated to the extent needed to determine serious potential problems.

"The estimated cost for work in this task is between \$0.8 and \$1.2 million.

"Sub-Task 7-4.3.4 Defected Fuel Element Behavior: Work will proceed as under Sub-Task 7-4.1.3.

"Under this Priority 1 sub-task, work is planned for FY 1969-FY 1973.

### "Task 7-4.4 Advanced Fuel Element

"If Tasks 7-4.2 and 7-4.3 do not achieve the desired improvement in performance of EBR-II, a new reference EBR-II fuel element will be designed and tested.

#### "Objective

"The objective of this task is to optimize the performance, reliability, and safety of EBR-II by developing a better design for a driver fuel element and establishing its operating limits.

"The new reference fuel element must incorporate a significantly higher design margin for operating temperature than the improved Mark II fuel element, thereby providing significantly greater assurance of safe performance.

### "Scope

"Work will be done on this task only if the efforts from Task 7-4.2 and 7-4.3 prove to be inadequate in achieving the task area objectives. In that case, other fuels that are being developed for commercial LMFBR applications will be reviewed to determine their applicability as driver fuels for EBR-II.

### "Plan of Action

"A new reference fuel system for EBR-II will be developed if work funded under Tasks 7-4.2 and 7-4.3 does not improve the operation of EBR-II to a satisfactory extent. If it is decided to proceed with this task, fuel systems will be evaluated first. After a reference system is chosen, a design will be developed and tested to establish whether it meets the objectives of this task. Subsequently, performance requirements will be assured by a statistical proof-testing program. The behavior of fuel elements under off-normal conditions and the consequences of operation with defective fuel elements will be investigated to the extent needed to determine whether serious problems can be expected.

"The new fuel must be fully developed and proved by about FY 1972. Therefore, it must not require a large advancement in technology to make it commercially available. It is unlikely that there will be enough fabrication experience with fuel types other than metals and oxides to satisfy this criterion.

"The estimated cost for work in this task is between \$2.0 and \$2.4 million.

"Sub-Task 7-4.4.5 Defected Fuel Element Behavior. Work will proceed as under Sub-Task 7-4.1.3.

"Under this Priority 1 sub-task, work is planned for FY 1970-FY 1974.

"Sub-Task 7-4.1.3 Defected Fuel Element Behavior: This sub-Task includes the following investigations:

## "(1) Instantaneous Effects of Fuel-Element Failure

"The behavior and immediate consequences of failure of intentionally defected Mark I series fuel elements will be investigated. The principal experimental variables will be location, size and type of defect, fuel-element design, and burnup. Tests may be performed in TREAT or in a loopsule in GETR or a similar thermal test reactor.

## "(2) Effects of Continued Operation of Defected Fuel Elements

"The effects of continued operation of defected fuel elements on their behavior and on the behavior of immediately adjacent fuel elements will be determined. The risk level of the initial tests precludes the use of EBR-II; therefore, the GETR loopsule with a seven-element capacity appears to be the best facility of near-term availability.

"Under this Priority 1 sub-task, work is planned for FY 1969-FY 1970.

[Note: These tests could be accomplished in an EBR-II Packaged Loop.]

## "Task 7-10.5 Failure Testing

### "Objective

"The objective of this task is determination of the failure threshold for fuel elements that are based on design concepts for commercial LMFBRs.

### "Scope

"The failure threshold of irradiated and unirradiated fuel elements will be identified through tests that accelerate failure by accentuating the conditions under which, and the mechanisms by which, failure can normally be expected. This task also is concerned with the operation of defected elements under normal and off-normal operating conditions to determine if these defects will propagate.

"This task is closely related to Task 7-6.7 and Task Area 7-5, and the Sodium Technology element.

### "Background Information

"The behavior of defective fuel elements under LMFBR conditions, especially after very high burnup, cannot be predicted on the basis of current knowledge; yet, defective fuel elements can be expected in any reactor no matter how elaborate the preventive measure. The number and severity of

defects that can be tolerated for commercial reactor fuel elements must be known. Defects in the cladding and in the sodium bond are of particular concern.

"The feasibility of operating fuel elements after cladding rupture has occurred is one of the most important considerations in choosing a fuel and fuel-element design for a power reactor. The inability of a defected element to sustain burnup safely after the time of failure results in serious economic consequences by:

"(1) Lowering the average core burnup by aborting the useful service of sound fuel elements in a core assembly that typically contains 100-500 fuel elements.

"(2) Requiring complex core instrumentation--possibly a sensor for each core assembly--to detect the failure and to locate, with minimal delay, the assembly containing the defective fuel element.

"(3) Magnifying greatly the risk (and undoubtedly the incidence) of unscheduled plant shutdowns, thus increasing the thermal cycles imposed on all components and complicating the management of the utility system that contains the plant.

"The possible rapid propagation of a failure is more serious since it is doubtful that the failure could be detected and remedial action (scram) taken in time to prevent gross contamination of the system.

"Investigations are needed to verify that during continued operation of a sodium-bonded oxide fuel element in which the bond has been lost, the fuel does not swell excessively, and excessive fuel does not escape from the fuel element. There is insufficient information to evaluate the other aspects of behavior of defected fuel elements--such as changes in fuel stoichiometry, contributions to fuel-cladding reactions, and fuel washout by the coolant.

"A major unresolved problem in the LMFBR Program is the performance of helium-bonded, mixed-oxide fuel in the presence of sodium coolant introduced through a cladding defect. A series of sodium-logging tests offers strong evidence that this phenomenon will not be a problem in the operation of defected fuel elements. However, this must be verified in high-burnup fuel elements irradiated in a fast flux because of the expected embrittled condition of the cladding and the need to lengthen the contact time between the fuel and sodium.

#### "Plan of Action

"The limiting conditions that would lead to termination of the useful life of a fuel element will be established for the fuel and fuel-element concepts of interest through experiments that aggravate the mechanisms

that lead to failure and thus accelerate it. The design margin of the fuel elements intended for specific missions will be established (as they are identified).

"The estimated cost for work in this task is between \$5.2 and \$7.8 million.

"Sub-Task 7-10.5.2 Defected-Fuel-Element Behavior Under Steady-State Conditions: Intentionally defected fuel elements will be irradiated under steady-state conditions in both thermal and fast flux test facilities to evaluate any tendency for failure propagation--either within the defected element or to adjacent elements. The character of the cladding defect (pinhole, slot, incipient crack) will be a test variable. The effects of normal thermal cycles will be investigated in sodium in the absence of irradiation. Exposures in these experiments will be continued to 50,000 MWd/T for evaluation.

"Under this Priority 1 sub-task, work is planned for FY 1970-FY 1975.

#### "Task 7-10.6 Performance Capability of Selected Fuel Systems

"After identification in FY 1973 of the most promising commercial fuel system(s), the scope of this task will include the engineering tests needed for the development of commercial LMFBR fuel-element designs. In-reactor investigating will include the correlation of design parameters with the results of operation under normal and off-normal conditions. For example, questions of spacer location and fuel-to-spacer gap will be examined, as will questions unique to a particular fuel system.

"The engineering-test environment should be typical of expected LMFBR conditions--e.g., fast-neutron flux, temperature, power rating, and sodium flow. These requirements will be tempered by limitations of existing facilities at the time these experiments are begun. The present capabilities of thermal-and fast-neutron irradiation facilities are summarized in Task Area 6-4 of the Core Design element of the Program Plan.

"The estimated cost for work in this task is between \$15 and \$23 million.

"Under this Priority 1 task, work will begin about 1974."

[Note: An EBR-II packaged loop could be used for such tests.]

### 3. Sodium Technology Program

The EBR-II packaged loop could provide a capability for experimental irradiations to support the work on materials compatibility conducted under Task Area 5-1 of the Sodium Technology Program Plan (Vol. 5).<sup>4</sup> Specifically, the loop could be used for experimental work in the tasks and subtasks enumerated below:

#### "Task 5-1.1 Metallic Mass Transfer

"Sub-Task 5-1.1.7 Radiation Effects on Mass Transfer: Possible mechanisms by which radiation could affect metallic solution or deposition rates on nonrefractory fuel cladding will be postulated. The possibilities for experimental measurement of the effect of radiation on metallic mass transfer by sodium will be evaluated. Preference will be given to the use of nonirradiated control specimens accompanying irradiation tests needed for other purposes. If it appears that a valid comparison can be obtained and is desirable, an experiment will be prepared for AEC approval.

"The experiment plan will be used to determine whether radiation significantly affects metallic mass transfer. The significance to target-plant operation of any effects observed will be evaluated.

[Note: The loop could provide means for investigating the effects of the radiation environment under specified, controlled coolant-chemistry conditions.]

#### "Task 5-1.4 Corrosion of Refractory Cladding Alloys

"Sub-Task 5-1.4.4 Effects of Radiation on Corrosion of Refractory Alloys: Possible mechanisms by which radiation could affect the corrosion rate of refractory-alloy fuel cladding, either directly or indirectly, will be postulated.

"The possibilities for experimental measurement of the effect of radiation on mass transfer of metals by sodium will be evaluated, preferably by devising nonirradiated control specimens to accompany irradiation tests needed for other purposes. If valid comparison appears possible and is desirable, an experiment plan will be prepared for AEC approval.

"The planned experiments will be performed, determining whether radiation significantly affects the corrosion rate and evaluating the significance to target-plant operation of any effects observed."

[Note: The loop could provide means for determining whether radiation significantly affects the corrosion rates of refractory alloys.]

## B. Experimental Program

For information, LMFBR core-design characteristics are summarized in Table II. Initial plans were that an EBR-II packaged loop would provide conditions to match many of the characteristics of the advanced-type LMFBR reactors. Because of its capabilities for higher temperatures, isolation from the reactor system, independent control of coolant chemistry, and failure containment, an EBR-II packaged loop would permit the performance of several types of experiments that could not normally be conducted in the reactor.

TABLE II. Summary of LMFBR Core-design Characteristics

Characteristics	EBR-II	FFTF	U.S. Demonstration Plant Studies	U.S. 1000-MWe Studies
Reactor power, MWt	62.5	400	600-1250	2345-2510
Coolant temperature, °F				
Inlet	700	550-800	700-800	720-979
Outlet	883	900-1200	950-1100	1028-1200
Max coolant velocity, ft/sec	24	30	25	14-40
Peak fuel temp with HCF, °F				
Cladding	1175	1075	1200-1300	1126-1400
Fuel	1320	4200	4200-4800	1440-4800
Avg specific power in core, kW/kg	304	360	800-1000	520-1680
Avg power density in core, kW/liter	735	375	300-650	105-695
Max linear power, kW/ft	13.7	14.4	14-17	10-44
Median fission energy, keV	350	-	~150	120-338
Peak neutron flux, n/cm <sup>2</sup> -sec x 10 <sup>-16</sup>	0.32	0.7	-	0.5-1.0
Fuel-element OD, in.	0.174	0.235	0.23-0.25	0.21-0.30
Fuel length, in.	14	36	33-50	36-50

The types of experiments that could be conducted in an EBR-II packaged loop are summarized below in the order of their present priority in the LMFBR Program Plan.

(1) "Off-nominal" testing of fuels and materials at higher power densities, temperatures, burnups, coolant impurity levels, etc., to establish feasible limits.

(2) Steady-state irradiation of fuel elements and materials to point of failure under future LMFBR conditions of temperature, power density, and coolant chemistry, to determine operating limits.

(3) Continued steady-state irradiation of fuel elements that have failed or have been given an intentional defect (referred to in this report as "defected" fuel elements), to determine consequences of failure.

- (4) Sodium-technology experiments
- (5) Vented-fuel experiments.

Because of the anticipated high demand for the limited loop facilities in EBR-II, very high burnups requiring long-term (years of) loop irradiation are not proposed. Such high burnups can be achieved by preirradiation, followed by use of a packaged loop for final testing. Safety-type experiments, such as loss-of-flow or overpower tests to investigate failure modes and failure propagation, are not proposed.

Most of the initial EBR-II loop experiments would be expected to involve uranium-plutonium oxides. Other fuel materials might be tested later. Cladding materials would probably include austenitic stainless steels at the start, to be followed by other alloys such as, but not limited to, nickel- and vanadium-base alloys.

### 1. Fuel-irradiation Capabilities

Fuel-irradiation experiments in an EBR-II packaged loop could include a number of fuel pins, consistent with the loop-design criteria. For example, in the available in-core experimental space, 7- or 19-pin clusters could contain elements with outside diameters of 0.360 and 0.235 in., respectively.

Calculated values of peak neutron flux and fission rate for several reactor power levels at the loop location (grid position 5N3) are listed in Table III for a 91-subassembly core with the present complete depleted-uranium blanket and with the future three-row nickel reflector.

TABLE III. Peak Neutron Fluxes and Fission Rates in  
EBR-II Packaged Loop in 91-subassembly Core

Reactor Power Level, MWt	Depleted-uranium Blanket				Nickel Reflector			
	Peak Neutron Flux, $n/cm^2\text{-sec} \times 10^{-15}$	Fission Rate, $\text{fis/g-sec} \times 10^{-13}$			Peak Neutron Flux, $n/cm^2\text{-sec} \times 10^{-15}$	Fission Rate, $\text{fis/g-sec} \times 10^{-13}$		
		$^{235}\text{U}$	$^{238}\text{U}$	$^{239}\text{Pu}$		$^{235}\text{U}$	$^{238}\text{U}$	$^{239}\text{Pu}$
1	0.04	0.0146	0.001	0.0175	0.041	0.0151	0.001	0.0181
50	2.0	0.73	0.05	0.88	2.1	0.76	0.05	0.91
62.5	2.5	0.91	0.07	1.10	2.6	0.94	0.07	1.13

The calculated neutron-energy distribution at the loop location is as follows:

Energy Level, MeV	Neutrons, %
Above 1.35	17
Above 0.83	29
Above 0.11	85
Below 0.11	15

The design loop-coolant temperature is 1200°F (650°C), but for special experiments, temperatures up to 1473°F (800°C) could be reached in the test section. Linear power densities of up to 40 kW/ft could be accommodated in an element of a seven-pin cluster.

Experimental instrumentation in the test section could measure such parameters as jacket and fuel temperatures, inlet and outlet coolant temperatures, coolant flow, coolant  $\Delta P$ , internal fuel-element pressures, and neutron flux. A cover-gas fission-product monitor would be provided to detect failure of experimental fuel.

Loop auxiliary equipment would include a plugging meter, a sampling station, and both cold and hot traps for monitoring and controlling the impurity content of the loop sodium.

The ranges of the planned initial testing capabilities of the EBR-II packaged loops and FFTF closed loops are compared in Table IV.

TABLE IV. Ranges of Initial Testing Capabilities  
of EBR-II and FFTF Sodium Loops

	EBR-II Packaged Loops	FFTF Closed Loops
Maximum number of loops	2	5
Maximum OD of test cross section, in.	1.7	2.75
Approximate test length above core, ft	8	15
Approximate test length below core, ft	1.5	2.5
Core height, in.	13.5	32
Fuel-element number and OD, in.	7-0.36 19-0.235	37-0.25
Total power, MWt	0.4	4.0
Max linear power per element, kW/ft	40	36
Peak flux, $n/cm^2$ -sec	$2.6 \times 10^{15}$	$7 \times 10^{15}$
Bulk $T_{in}$ , °F	700-1200	500-1100
Bulk $T_{out}$ , °F	700-1475	800-1400
Bulk $\Delta T$ , °F	0-325	20-400
Max test section $\Delta P$ , psi	40	100
Flow, gpm	10-70	70-400
Max velocity, ft/sec	40	-

## 2. Off-nominal Testing

A high-priority item in the LMFBR Program Plan is the testing of fuels and materials at higher-than-normal power densities, temperatures, and burnups, or in sodium with off-normal impurity contents, to determine the failure limits.

In a fuel element, higher burnup, higher power density, and higher temperature all increase the risk of cladding failure with consequent fission-product release and perhaps fuel movement. Because of the increased risk of failure, such tests are not presently allowed in EBR-II. The packaged loop in EBR-II would be particularly well suited to carrying out irradiation experiments to determine the burnup and temperature limits of a fuel material or fuel-element design under off-normal conditions. Fuel irradiations could be carried to failure without contamination of the reactor's primary sodium system by plutonium and fission products. A fission-product monitor, which was part of the loop instrumentation, would be used to determine the onset of failure. High burnups could be achieved by using preirradiated fuel.

The packaged loop would enable measurement of the effective thermal conductivity of experimental fuel elements while they were operating under conditions near those that would cause cladding failure. More importantly, the central fuel temperature could be measured continuously while the element was at full power during such events as cladding failure, sodium logging, and bond loss.

## 3. Testing to Failure

Another high-priority program is the steady-state irradiation of fuels and materials to the point of failure under future LMFBR conditions of temperature, power density, and coolant density. The objective of these tests is to establish operating limits and allowable fuel burnups on specific fuel designs and materials.

Tests in which fuel-element failure is gradually and deliberately induced under controlled conditions to determine the weakest point of the fuel element are an important part of the fuel-development program. The failure tests under the irradiation program will take two general forms: (a) defect tests, in which a cladding penetration is produced in a controlled manner during irradiation; and (b) failure tests, in which extensive cladding failure is promoted by intentional overheating, overstress, excessive burnup, etc. Preirradiation of test elements and final irradiation in a packaged loop could be used to attain failure due to high burnup.

Information on the mode of failure is needed to permit analysis of the reactor hazards associated with fuel-element failure. Although a substantial effort has been, and is being devoted to the consequences of various types of fuel failure, more convincing information will result from further studies of actual failures. Both driver fuel and prototypical oxide fuels could be irradiated up to and including the failure point. Postirradiation examination of the failed specimens would provide valuable information on the actual hazards implications of failure and provide a better understanding of the responses of fission-product-monitoring systems to various types of failures. Such information is vital to the fuel-failure diagnostics program. In the packaged loop, LMFBR conditions of coolant, fuel and cladding temperatures, power densities, and sodium impurities could be achieved. Testing to failure could be achieved without contaminating the reactor system. Containment of the failed fuel within the loop test section would be ensured by using suitable barriers.

The high design pressure of the loop would permit containment of fuel failures and other postulated accidents with a minimum risk of damage to the reactor and hazard to personnel. The loop fission-product monitor, experimental temperature measurements, and analyses of sodium samples could be used to detect the failure.

#### 4. Operation with Failed or Defected Fuel

Continued steady-state irradiation of failed or defected fuel elements with the objective of determining the consequences of fuel failure is an important part of the LMFBR Program Plan. Information on fuel movement, slumping, melting, cracking, diffusion, swelling, void formation, fuel-cladding reactions, and propagation of failure to adjacent elements is needed.

Tests to obtain the above information could be carried out in an EBR-II packaged loop. The high-pressure loop-containment design and the shielding of components external to the EBR-II primary tank would permit operation of the loop with failed or defected fuel elements with minimum risk to the reactor or hazards to personnel.

One experiment of interest would be the reirradiation of special driver elements, which have been "recast" in flat-topped TREAT transients. As past and current TREAT tests indicate, the consequences of a loss-of-bond type of failure in an EBR-II driver element are not severe. It is hoped that eventually such failures can be tolerated without shutting down the reactor. To prove the feasibility of operating with such a failure, however, the effects of prolonged running with a low-burnup fuel material will need to be investigated. High-burnup material will have been

discharged earlier, but low-burnup material will continue to swell after recasting and may produce enough stress to split the jacket. The packaged loop could be used to investigate the attendant hazards of such a secondary failure.

Similarly, any special hazard associated with the failure of unencapsulated oxide-fuel elements could be evaluated from tests conducted in the packaged loop. Such experiments would deal with gas-blanketing phenomena and any sodium-fuel interaction resulting from the continued operation with a jacket failure.

## 5. Sodium Technology

Although fueled experiments have first priority, the EBR-II packaged loop could be used for investigating interactions of candidate LMFBR cladding and structural materials with sodium and its impurities, and the resulting effects upon the mechanical properties of materials, in a fast-reactor radiation environment. The loop design includes provisions for establishing and maintaining specified concentrations of particular impurities in the loop sodium for experimental purposes.

At present, no materials have been conclusively demonstrated to be acceptable for use as fuel cladding or for structural purposes in an LMFBR designed for long-time reliable operation with a bulk-exit sodium temperature of 1200°F. On the basis of considerable laboratory testing, stainless steels show promise for this application. Other alloy systems are being evaluated for "backup" to stainless steels.

The sodium-technology work in EBR-II packaged loops might logically be considered in the following steps:

- a. Establish whether a nuclear-reactor environment has a significant effect on the compatibility of structural materials with sodium, as related to LMFBR performance.
- b. If there is such an effect, establish a program to determine its salient features, and, as permitted by time and the facilities available, describe the effect quantitatively so that the information may be as useful as possible for LMFBR design.

The extent and degree of mass transfer (of constituents of the materials, and of interstitial elements), and therefore of possibly important property changes, may be influenced by small changes in concentrations of certain impurities in the sodium and by the magnitudes of impurity sources. Synergistic effects (impurity-impurity and impurity-irradiation) may also be important.

Since the exact conditions of LMFBR operation will not be known for an appreciable time, the behavior of materials exposed to sodium in a fast-neutron environment must be determined over a range of sodium conditions not practical to achieve in the primary circuit of EBR-II. These conditions can be achieved in a packaged loop (containing in-core and out-of-core test sections) by using sodium supply separate from that of EBR-II. Comparison of materials behavior in EBR-II coolant and in the loop coolant would allow development of insight regarding possible synergistic effects of variables in the complete reactor environment.

It will be possible to perform in-reactor sodium-corrosion experiments in the range of 500-650°C (932-1202°F) in the proposed EBR-II packaged loops. With a specially designed test section in which the hot outlet coolant is mixed with bypassed inlet coolant, temperatures up to 800°C (1472°F) could be achieved. Experiments to determine the effects of coolant velocity could be performed with velocities of 4-30 ft/sec.

Sodium chemistry would be controllable by a cold trap, a hot trap, solution getters, and suitable injection of desired additives. With respect to sodium composition, elements of interest include oxygen, carbon, nitrogen, hydrogen, and metals.

The alloys to be irradiated would consist principally of those suitable for fuel-cladding and core-structure applications. These alloys include stainless steel, high-strength iron- and nickel-base alloys, and refractory-metal alloys.

The investigation of fission-product behavior in sodium systems is not a primary objective of this loop. However, significant information concerning such behavior could be expected as a by-product of fuel-element tests with fission-product release.

Significant information concerning possible synergistic effects of radiation and coolant chemistry upon corrosion rates of cladding alloys could also be expected as a by-product of fuel-element tests under specified and controlled coolant-chemistry conditions for which out-of-pile corrosion data exist.

## 6. Vented-fuel-element Tests

Although it is a low-priority item in the LMFBR Program Plan, development work on advanced fuel concepts such as vented fuel elements is proposed for the future. The irradiation of prototype vented fuel is needed to gain information on (a) the rate of fission-product release to the primary coolant, (b) the effect of vent-hole size on fission-product release, and (c) the extent and effect of sodium logging of fuel elements. Knowledge of the

effects of total burnup, burnup rates, temperature, fission-product migration, and coolant flow on the adequacy of the proposed vent designs is also needed.

Vented-fuel-element concepts and techniques could be evaluated in an EBR-II packaged loop. A loop would provide an isolated environment for such tests, so that fission-product-release to the loop coolant could be monitored with high sensitivity. The fission-product retention characteristics of the sodium could be studied under a variety of conditions.

## IV. SAFETY CONSIDERATIONS

### A. Introduction

The safety aspects of design, fabrication, installation, and operation of the EBR-II packaged loop will be the principal concern in any future engineering design effort. The safety design criteria (SDC) formulated in the feasibility study will be enlarged to encompass the complexity of an operating loop in the EBR-II reactor core. The SDC will then be used in various trade-off studies to optimize the utilization of the packaged loop. These trade-off studies will emphasize design features, fabrication difficulties, operational variables, and plant safety requirements.

Upon the completion of a detailed design, fault-tree analysis<sup>5</sup> will be used to study each subsystem associated with the loop and the influence that faults would have on surrounding equipment and reactor availability. These fault-tree analyses will be followed by a preliminary hazards analysis of the loop installation.

The paragraphs below list some of the principal safety considerations that will be used in the engineering trade-off studies. This list will be enlarged, as necessary, to conform to the high standards of safety presently applied at EBR-II.

### B. Packaged-loop Safety Considerations

This discussion is subdivided into five categories, similar to those published by the Atomic Energy Commission in "General Design Criteria for Nuclear Power Plant Construction Permits."<sup>6</sup> The general intent of those criteria applicable to the quality assurance and safety design of the packaged loop in EBR-II have been interpreted and are listed below. The five categories considered are:

- (1) General criteria
- (2) System containment
- (3) Instrumentation
- (4) Safety features
- (5) Radioactivity release.

#### 1. General Criteria

a. Those systems and components of the packaged loop that are essential to the prevention of accidents affecting plant safety or to the mitigation of their consequences shall be identified, then designed, fabricated, and erected to standards of quality that reflect the importance of the safety function to be performed. Recognized codes and standards of design,

material, and fabrication and inspection shall be identified. These standards shall be supplemented or modified as necessary to ensure the safety of a packaged loop in EBR-II.

b. The packaged loop shall be designed to (a) minimize the probability of events such as sodium fires and sodium interactions with molten fuel and (b) minimize the potential effects of such events to EBR-II plant safety.

## 2. System Containment

a. The in-pile containment tube shall be designed and constructed to possess an exceedingly low probability of gross rupture or significant leakage throughout its design operating lifetime.

b. The in-pile containment tube shall be capable of accommodating, without rupture (and with only limited allowance for energy absorption through plastic deformation), the static and dynamic loads imposed on any containment component as a result of an inadvertent and sudden energy release to the loop coolant (i.e., sodium-molten fuel interaction). A preliminary accident analysis has been made to establish a basis for a system design pressure (see Section IV.D).

c. The permanent piping system and auxiliary components shall have provisions for periodic inspection, testing, and surveillance by appropriate means to assess the structural and leak-tight integrity of the containment components during their service life.

## 3. Instrumentation

a. Instrumentation and control for the loop shall be provided, as required, to monitor and maintain variables within prescribed operating ranges.

b. Means shall be provided for monitoring the loop system to detect coolant leakage.

c. The loop protective systems shall be designed for high functional reliability and for in-service testability in accord with the importance of the safety functions they are to perform.

d. Enough redundancy or backup shall be designed into the protective system to ensure that no single failure or removal from service of any loop component or channel of a system will result in loss of the protective function. Where feasible, the redundancy shall normally include two channels of protection for each protective function to be served. Different principles shall be used, where practical, to achieve true independence of redundant instrumentation components.

e. Emergency sources of power shall be provided for the protective systems.

f. Means shall be included for testing protective systems while the loop is operational to demonstrate that no failure or loss of redundancy has occurred.

#### 4. Safety Features

a. Engineered safety features shall be provided in the loop to back up the safety provided by the loop design, the auxiliary systems, and the protective systems. As a minimum, such engineered safety features shall be designed to minimize the effect of the design-basis accident involving the loop system.

b. Safeguards shall be provided for the piping system and auxiliary-component protective instrumentation to guard against dynamic effects and missiles that might result from plant-equipment failure.

c. Two loop cooling systems, one for normal operation and one for emergency shutdown conditions, shall be provided.

d. The loop containment structure, including access openings and penetrations and any necessary containment systems, shall be designed so that the structure can accommodate the largest credible energy release that would follow a loss of coolant flow in the loop (without exceeding design pressures and temperatures and including a considerable margin for the effects of sodium-molten-fuel interaction).

#### 5. Radioactive Releases

The loop design shall include those means necessary to maintain control over radioactive releases, whether gaseous or solid. Appropriate holdup capabilities shall be provided for retention of gaseous or solid effluents. For all cases, the design for radioactivity control shall be based on (a) the 10-CFR-20 requirements for normal operation and for any transient situation that might reasonably be anticipated to occur, and (b) the 10-CFR-100 dosage-level guidelines for potential release following an occurrence of exceedingly low probability.

#### C. Summary of Safety Considerations

The above considerations reflect the principal safety-problem areas in the packaged-loop conceptual design. The Preliminary Safety Analysis Report will focus on a conceptual loop design and analyze potential loop-system faults that could lead to abnormal loop operating conditions. Primary interactions between the loop and the main reactor core and

process system will be studied to identify feedback loops that could affect reactor operations. A final Safety Analysis Report will summarize the principal packaged-loop safety-design criteria, the system fault tree and safety analysis, and the design-basis accident for the loop-containment barrier.

#### D. Preliminary Accident Analysis

##### 1. Introduction

To establish a basis for the loop design pressure, a preliminary look has been taken at what might be a maximum accident in the EBR-II packaged loop. The analysis is based on examining the capability of the loop to withstand an experiment in which 132 cc of oxide fuel (1320 g of  $\text{UO}_2$  in 19 pins of 0.235-in. OD) is heated uniformly to 3450°K (5782°F). The fuel is contained in Type 316 stainless steel cladding. (See Fig. 5, Section F-F on p. 53.)

The analysis is similar to part of the study made for the Mark-II TREAT integral sodium loop<sup>7</sup> and assumes a failure mechanism that has been treated by Hicks and Menzies.<sup>8</sup> The mechanism assumes that the primary energy release is concentrated in the fuel and that the sodium coolant remains relatively cool. This primary energy release may disperse the fuel through the coolant so that sodium boiling and vaporization, in a secondary energy release, could produce pressures of explosive violence. The primary energy release in a large fast power reactor, resulting in the spewing of  $\text{UO}_2$  fuel into the sodium coolant, will be relatively small.<sup>8,9</sup> The cause of such an energy release could be a loss-of-flow or reactivity accident.

The mixing of hot fuel with the sodium coolant, where the expanding sodium does work on the environment, is considered a maximum accident. The energy and pressures resulting from this secondary energy release will be compared with the energy and pressures resulting from a quantity of chemical explosives that will rupture the inner hexagonal tube of the loop.

The deformation of a cylinder from explosive shock waves is relatively large.<sup>10</sup> If the outer containment tube of the loop suffered such a deformation, the loop thimble and adjacent fuel assemblies would probably be damaged. Such damage could make it difficult to remove the loop from the reactor. The energy available from the accident is primarily absorbed by the inner hexagonal tube, which contains the fuel pins. Therefore, the outer containment tube, which is the loop boundary, should remain intact.

It is assumed that the inner hexagonal tube absorbs very little energy as it becomes circular in shape and that it acts as a cylinder of

uniform thickness thereafter. There is only small clearance between the outer tube and the thimble to allow a flow of primary reactor coolant for removal of heat from the loop. However, there should be negligible deformation of the outer tube.

## 2. Loop Pressure Capabilities

a. Steady-state Code Rating. Section III of the ASME Boiler and Pressure Vessel Code for Nuclear Vessels does not give allowable design stresses for temperature above 1000°F. Special Case 1331-4, approved August 15, 1967, directs the use of Section I of the code at higher temperatures. Design pressures for different sections of the loop are shown in Table V. These ratings are based on the following equation (from Appendix I of Section III of the ASME Boiler and Pressure Vessel Code for Nuclear Vessels):

$$P = \frac{tS_m}{R + 0.5t},$$

where

P = design pressure,

t = tube thickness = 0.100 in.,

S<sub>m</sub> = allowable stress,

and

R = internal radii of tubes: Core section of outer tube = 0.98 in.;  
Core section of inner hexagonal tube = 0.69 in. (equivalent radius); Outer tube above core = 1.15 in.

TABLE V. ASME Code Pressure Rating

Loop Boundary Section	Max Operating Temp, °F	Design Stress		Design Pressure, psi
		psi	Ref.	
Core	1000	14,130	Section III 90% yield	1375
Above Core	1200	6,950	Section I	580
Pump	1000	14,130	Section III 90% yield	489

The above stress values, which are used for the code rating, are conservative for the packaged loop because:

(1) The code is based on an operating life of 20 years; a packaged-loop pressure tube is expected to have an operating life of 2 years or less.

(2) As a basis for the allowable stress, the code considers the normal operating temperature and pressure with allowance for pressure and temperature transients. The packaged loop will operate at design temperature or less, but its normal operating pressure will be only 100 psi or less. The pressure requirement for the loop is based on a postulated maximum accident, which would be a once-in-a-lifetime occurrence.

Heat-transfer studies have shown sodium coolant temperatures in the packaged loop, as listed in Table VIII (p. 82). This information has been used in determining maximum operating temperatures for the outer tube and the inner hexagonal tube.

b. Static Bursting Pressure,  $P_{SB}$ , for Inner Hexagonal Tube.  
Use of the following equation for a cylinder by Boers:<sup>11</sup>

$$P_{SB} = \frac{2\sigma_y}{\sqrt{3}} \ln \left[ \frac{R_o}{R_i} \left( 2 - \frac{\sigma_y}{\sigma_u} \right) \right],$$

where

$\sigma_y$  = yield stress = 20,300 psi,

$\sigma_u$  = ultimate stress = 65,500 psi,

$R_o$  = external radius = 0.79 in.,

and

$R_i$  = internal radius = 0.69 in.,

results in a static bursting pressure of 15,400 psi for the inner hexagonal tube. The above stress values are taken from Ref. 12 for 1100°F, which is the maximum operating temperature of the inner hexagonal tube.

c. Amount of Chemical Explosives Required to Rupture Inner Hexagonal Tube. Although no exact method is known for calculating the explosives needed for rupturing a cylinder, two methods (each based on experimental data) are used. The first method is based on testing performed at the NRTS.<sup>13</sup> The second method was developed at the Naval Ordnance Laboratory.<sup>10</sup>

The material properties of Type 316 stainless steel at 1100°F are as follows:<sup>12</sup>

Yield stress	20,300 psi
Ultimate stress	65,500 psi
Elongation	43.5% (2 in.)
Reduction in area	60.5%
Modulus of elasticity	$21.9 \times 10^6$ psi
Density	500 lb/ft <sup>3</sup> .

The first method is based on the equation

$$\Delta = 169 \left( \frac{\omega}{\sigma_y R_i t} \right)^{0.82},$$

where

$\omega$  = explosive charge, grains/ft

$R_i$  = internal radius = 0.69 in.,

$t$  = cylinder wall thickness = 0.10 in.,

$\Delta$  = permanent diametral deformation (% elongation/3) = 14.5.

Rearranging terms gives

$$\omega = \sigma_y R_i t \left( \frac{\Delta}{169} \right)^{1.22}$$

= 70 grains PETN (Primacord) per foot of charge.

The tests of Ref. 13 used small-diameter Schedule 80 cylinders 28 and 36 in. long. The Primacord explosive was centered through the entire length of the cylinder. Taking the 13.5-in. fuel length of the packaged loop into account and converting to equivalent energy gives

$$\begin{aligned} \text{Energy} &= \frac{70 \text{ grains PETN/ft}}{7000 \text{ grains/lb}} \times \frac{6.5 \times 10^5 \text{ cal}}{1 \text{ lb PETN}} \times \frac{13.5 \text{ in.}}{12 \text{ in./ft}} \\ &= 7300 \text{ cal.} \end{aligned}$$

The second method is based on the equation

$$W = \left[ \frac{1.407 \sigma_t \epsilon (3.41 + 0.117 R_i / h_0) (R_e^2 - R_i^2)^{1.85}}{10^5 \omega^{-0.85} (1.47 + 0.0373 R_i / h_0)^{0.15} R_i^{0.15}} \right]^{0.811},$$

where

$$\sigma_t = \sigma_y + \left[ \frac{\sigma_u(1 + \epsilon_u) - \sigma_y}{\epsilon_u} \right] \epsilon = 35,400 \text{ psi},$$

using

$$\epsilon_u = \frac{\sigma_u}{E} \text{ and } \epsilon = \frac{\epsilon_u}{3},$$

$\epsilon$  = permissible strain at temp = 0.00088 in./in.,

$\epsilon_u$  = conventional ultimate strain at temp = 0.00264 in./in.,

$R_e$  = external cylinder radius = 0.79 in.,

$R_i$  = internal cylinder radius = 0.69 in.,

$h_0$  = wall thickness = 0.10 in.,

$\omega$  = density of cylinder material = 500 lb/ft<sup>3</sup>,

$E$  = modulus of elasticity =  $21.9 \times 10^6$  psi,

and

$W$  = charge weight contained = 0.0252 lb of TNT.

The equivalent energy is

$$\text{Energy} = 0.0252 \text{ lb TNT} \times \frac{5 \times 10^5 \text{ cal}}{1 \text{ lb TNT}} = 12,600 \text{ cal}.$$

This energy value is about 73% greater than the value calculated by the first method. This agreement is not too good, considering that the lower value is for rupture while the higher value is for containment but impending rupture. The discrepancy can probably be attributed to (a) differences in test configurations in the experiments in Refs. 10 and 13, and (b) the fact that the inner hexagonal tube is smaller in equivalent diameter than any of the specimens tested. Since the energy values are of the same order of magnitude, an average energy value from the two methods above of 9950 cal in the form of a shock wave will be considered the energy required to rupture the inner hexagonal tube.

A superhigh-alloy steel, Carpenter A-286, has also been considered in this analysis. Although ultimate and yield stresses at temperature are many times those of the referenced design material (Type 316 stainless steel), there is little difference in the energy value for rupture, because of the elongation properties of the Carpenter A-286 steel.

### 3. Upper Limit of Work Available in Shock Wave from Vaporization and Expansion of the Sodium Coolant

Hicks and Menzies<sup>8</sup> have presented results from calculations for an initial oxide-fuel temperature of 3450°K and an initial sodium temperature of 1150°K. This UO<sub>2</sub> fuel temperature is the estimated boiling point at atmospheric pressure. Assumptions made in their calculations are:

a. The sodium vapor arises from instantaneous complete homogenous mixing of the oxide fuel with the sodium coolant. The mixing takes place at constant volume, and the temperature equilibrium is obtained before any expansion takes place.

b. The mixture then begins to expand adiabatically, remaining in thermodynamic equilibrium throughout the expansion.

c. Liquids are incompressible and of negligible specific volume when compared with the vapor phase. Sodium vapor behaves as a perfect gas, and specific and latent heats are constant.

Reference 8 states that the following equations, based on a simple approximate two-phase equation of state for sodium, give a reasonably adequate representation of the correct results:

$$\frac{Lx}{T_1} = (mS_\ell + S_f) \ln \frac{T_m}{T_1},$$

$$T_m = \frac{mS_\ell T_c + S_f T_f}{mS_\ell + S_f},$$

and

$$W = (mS_\ell + S_f)(T_m - T_1) - x(L - RT_1),$$

where

$m$  = 0.07 g of sodium per g of UO<sub>2</sub>,

$x$  = 0.07 g of sodium vapor per g of UO<sub>2</sub>,

(it is assumed that  $x = m$  or that all sodium within the inner hexagonal tube of the core section vaporizes),

$R$  = gas constant,

$L$  = latent heat of evaporation = 4000 J/g,

$T_m$  = initial equilibrium temperature,

$T_1$  = required end temperature where  $x = m$ ,

$S_\ell$  = specific heat of sodium = 1.2 J/g-°C,

$S_f$  = specific heat of  $\text{UO}_2$  =  $0.3 \text{ J/g}^\circ\text{C}$ ,

$T_c$  = initial sodium temperature =  $1150^\circ\text{K}$  ( $1611^\circ\text{F}$ ),

and

$T_f$  = initial fuel temperature =  $3450^\circ\text{K}$  ( $5815^\circ\text{F}$ ).

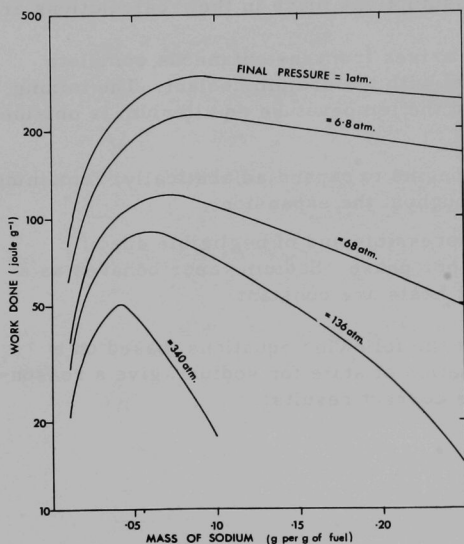


Fig. 1. Work Done in a SPERT-type Incident with Sodium and  $\text{UO}_2$ . ANL Neg. No. ID-103-K5901.

(823 psi). The use of the loop design temperature,  $1200^\circ\text{F}$ , in place of  $1611^\circ\text{F}$  results in only slightly lower energy values.

If the Ref. 8 calculations were extended to cover a wider range of initial sodium temperatures, one could establish a condition that is probably more realistic. The new condition is that further heat transfer from the fuel is prevented by vapor blanketing as the hot sodium expands. This assumption deletes the fuel-energy term,  $S_f$ , from the above equations and results in pressure-wave energy-release values of about one-fourth of those calculated by Hicks and Menzies. The use of this new condition results in the following total amount of pressure-wave energy available for doing work:

$$\text{Energy} = \frac{1}{4} \cdot \frac{130 \text{ J}}{\text{g of } \text{UO}_2} (1320 \text{ g of } \text{UO}_2) \frac{\text{cal}}{4.187 \text{ J}} = 10,300 \text{ cal.}$$

This value is close to the amount of explosive energy, 9950 cal, considered necessary to rupture the inner hexagonal tube.

Hicks and Menzies showed that the work potential varied with the sodium-to-fuel ratio and that a maximum existed. The maximum corresponded to a ratio that produced complete evaporation of the liquid with insignificant superheating of the vapor. As shown by Fig. 1,<sup>8</sup> the ratio of sodium to fuel in the 19-pin loop configuration, 0.07 g of Na per gram of  $\text{UO}_2$ , gives near a maximum work potential if one assumes a final pressure of 136 atm (2000 psi). Entering Table VI<sup>8</sup> with an  $m$  value of 0.08 and assuming  $x = m$  results in a pressure-wave energy available for doing work of 130 J/g of  $\text{UO}_2$ . Table VI also shows that when all the loop sodium initially in the core section is vaporized, the terminal temperature and pressure are  $1883^\circ\text{K}$  and 56 atm

TABLE VI. Adiabats of a Mixture of Sodium and  $\text{UO}_2^{(8)}$ 

Mass of Sodium per Unit Mass $\text{UO}_2$	Temp, $^{\circ}\text{K}$	Pressure, atm	Mass of Sodium Vapor per Unit Mass $\text{UO}_2$	Volume of Sodium Vapor per Unit Mass $\text{UO}_2$ , $\text{cm}^3/\text{g}$	Work Done in Expansion per Unit Mass $\text{UO}_2$ , J/g	Mass of Sodium per Unit Mass $\text{UO}_2$	Temp, $^{\circ}\text{K}$	Pressure, atm	Mass of Sodium Vapor per Unit Mass $\text{UO}_2$	Volume of Sodium Vapor per Unit Mass $\text{UO}_2$ , $\text{cm}^3/\text{g}$	Work Done in Expansion per Unit Mass $\text{UO}_2$ , J/g
0.01	3362	930	0	0	0	0.02	3280	850	0	0	0
	3231	805	0.01	0.132	11.6		3023	624	0.02	0.319	23.4
	3201	340		0.310	20.8		2985	340		0.577	35.2
	3170	136		0.766	30.2		2929	136		1.416	52.7
	3146	68		1.521	37.6		2886	68		2.791	66.0
	3069	6.8		14.84	61.1		2750	6.8		26.59	108.4
	3006	1.0	0.01	98.9	80.4		2697	1.0	0.02	177.4	148.3
0.03	3204	780	0	0	0	0.04	3133	717	0	0	0
	2823	470	0.03	0.593	36.2		2631	345	0.04	1.004	49.8
	2795	340		0.811	45.1		2630	340		1.017	50.1
	2718	136		1.971	69.5		2536	136		2.452	80.5
	2661	68		3.860	87.6		2467	68		4.771	102.7
	2480	6.8		35.97	144.9		2252	6.8		43.55	172.1
	2339	1.0	0.03	230.8	189.6		2086	1.0	0.04	274.4	225.7
0.06	3005	608	0	0	0	0.08	2892	520	0	0	0
	2623	340	0.0332	0.842	38.3		2623	340	0.0253	0.642	27.4
	2258	163	0.06	2.741	83.0		2185	136	0.0606	3.201	81.7
	2235	136		3.242	90.7		1940	68	0.0767	7.504	119.8
	2150	68		6.237	119.1		1883	56	0.08	8.796	129.8
	1887	6.8		54.74	206.9		1618	6.8		62.58	228.4
	1694	1.0	0.06	334.3	271.4		1410	1.0	0.08	371.0	305.8
0.10	2793	450	0	0	0	0.15	2588	320	0	0	0
	2623	340	0.0173	0.439	17.3		2185	136	0.0444	2.345	48.2
	2185	136	0.0563	2.974	71.2		1940	68	0.0671	6.294	86.0
	1940	68	0.0742	6.960	109.4		1414	6.8	0.1026	70.14	201.5
	1444	8.1	0.10	58.67	214.7		1153	1.0	0.1119	424.4	284.2
	1423	6.8		68.80	222.9						
	1208	1.0	0.10	397.3	306.7						
0.20	2428	235	0	0	0	0.25	2300	179	0	0	0
	2185	136	0.0311	1.643	29.5		2185	136	0.0168	0.887	14.0
	1940	68	0.0587	5.506	66.7		1940	68	0.0497	4.662	49.3
	1414	6.8	0.1032	70.55	183.4		1414	6.8	0.1032	70.55	167.4
	1153	1.0	0.1159	439.5	269.4		1153	1.0	0.1194	452.8	256.5

This preliminary study considers only order-of-magnitude values. Reference 8 indicates that the problem is too complex to allow conclusive calculations to be made. Among the uncertainties involved are:

- The degree of intermixing of the  $\text{UO}_2$  fuel with the sodium coolant. The energy calculation above should be conservative, because instantaneous complete homogeneous mixing is assumed.
- The degree of heat transfer during the adiabatic expansion process. The less conservative Edwards assumption has been used.
- The properties of sodium at the higher temperatures and near the critical pressure region.
- The possibility of some unknown critical phenomenon which is not included in the simplified equations.

The outer tube is the same thickness (0.100 in.) as the inner hexagonal tube, but, because of its larger radius, will contain higher explosive pressures, according to equations in Refs. 10 and 13. The thickness of the inner hexagonal tube could be increased slightly to give a higher pressure rating for the lower sodium-coolant flowrates. However, there should be little, if any, permanent deformation of the outer tube with an

inner-hexagonal-tube thickness of 0.100 in. The 1375-psig design pressure for the test section appears adequate. The pressure pulse from a postulated accident has only negligible effect on the remainder of the loop and does not affect the design pressure required.

## V. DESCRIPTION OF FACILITY

### A. Pump Concepts

The major components of the EBR-II packaged loop are shown in the schematic flow diagrams of Figs. 2-4. These figures present two concepts, whose main difference is in the location of the loop pump. Figures 2 and 3 show the pump located within the in-pile tube inside the primary-tank shielding. The in-pile containment tube is defined as the small-diameter container that fits through a reactor control-rod-drive opening in the small rotating plug and runs from the top of the plug to the bottom of the reactor core. The advantages of this internal pump arrangement are: (1) All radioactive sodium in the loop primary system can remain within the reactor's primary-tank shielding, and (2) the congestion caused by excessive auxiliary equipment on top of the rotating plug is eliminated. Figure 2 shows sampling, purification, and plugging-meter equipment which can be connected to the system for special experiments. The external shielded package is small compared to the external pump arrangement, where the pump and sodium containers are located externally. When the special equipment is not used, this internal pump installation should result in no more congestion or interference than the instrumented-subassembly installation. The primary difference between the designs of Figs. 2 and 3 is that a concentric-tube heat exchanger is used in Fig. 2 and a multitube heat exchanger in Fig. 3.

Figure 4 shows the pump located within a shielded equipment package outside the primary-tank shielding. Disadvantages of this external pump arrangement are: (1) the large mass required to shield the activated loop sodium and the fission products in case of a fuel-element rupture; (2) the congestion on top of the reactor, limiting access to other equipment; (3) the requirement for elevating the heavy shielded equipment package 6 ft to allow refueling, or the use of flexible sodium tubing so the equipment package can remain stationary; and (4) the requirement for opening the loop sodium system when disconnecting the in-pile tube from the equipment package during removal of a completed experiment.

Figures 5-7 present further details of the concepts of Figs. 2-4. Figures 5 and 6 show internal EM pump arrangements with concentric-tube and multitube heat exchangers, respectively. Figure 7 shows an external pump arrangement with a concentric-tube heat exchanger.

### B. Test Section

#### 1. Fueled Experiments

The hexagonal test section, which extends through the core and axial blanket regions, as shown in Figs. 5 (Section F-F), 6 (Section K-K), and 7 (Section B-B), is about 1.7 in. in outer dimension (across the flats)

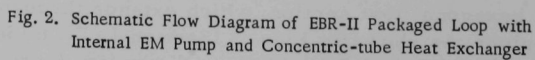


Fig. 2. Schematic Flow Diagram of EBR-II Packaged Loop with Internal EM Pump and Concentric-tube Heat Exchanger

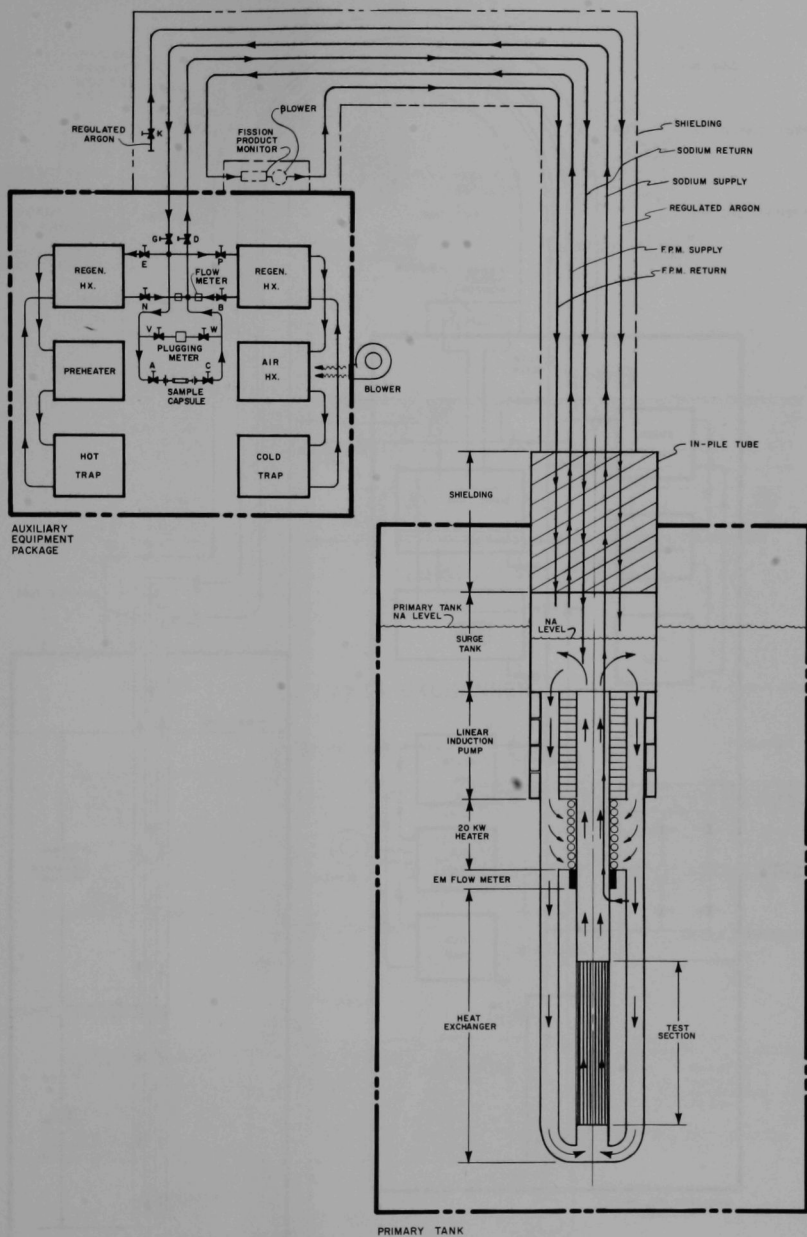


Fig. 3. Schematic Flow Diagram of EBR-II Packaged Loop with Internal EM Pump and Multitube Heat Exchanger

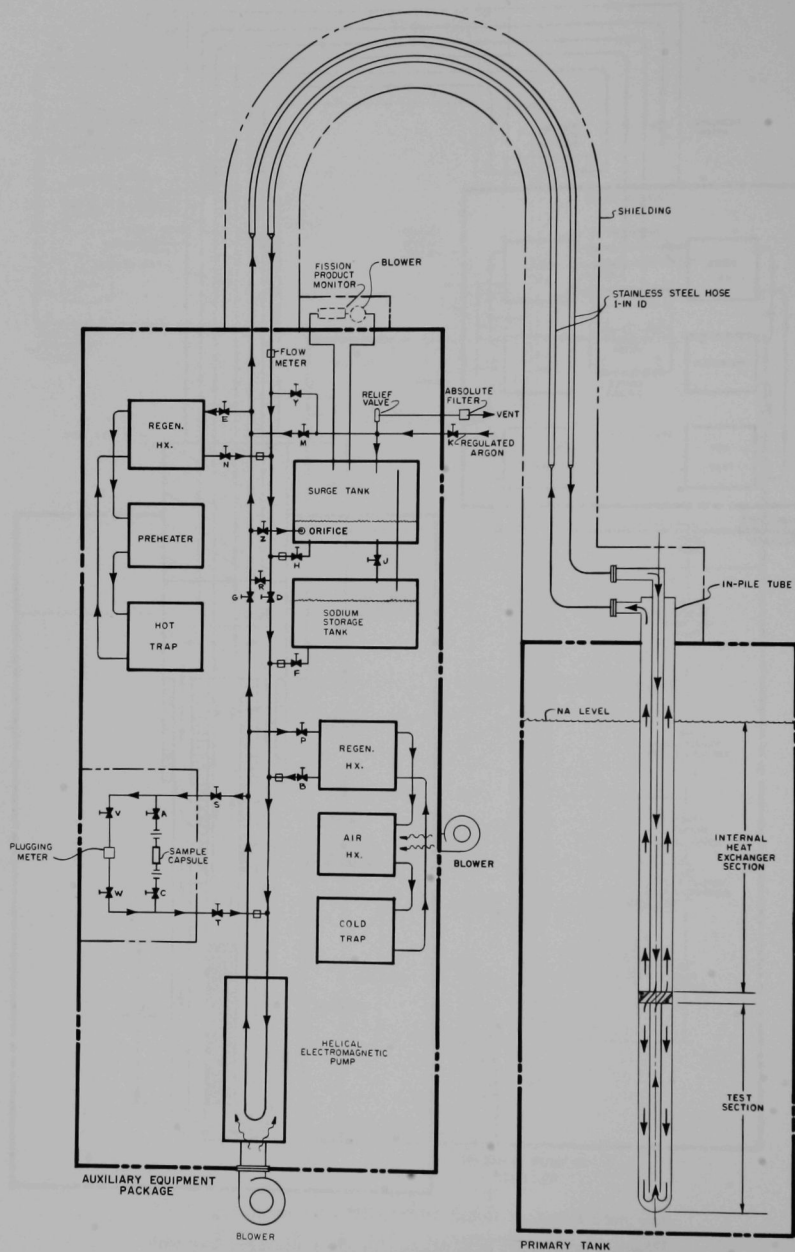


Fig. 4. Schematic Flow Diagram of EBR-II Packaged Loop with External Pump

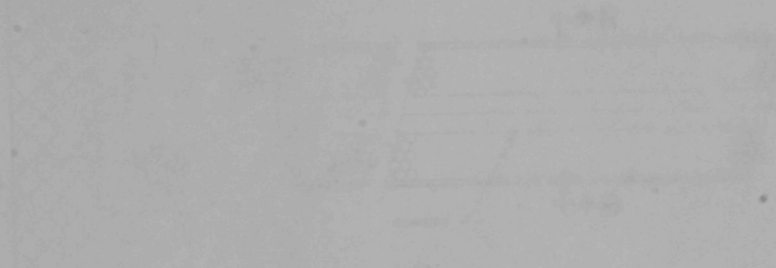


FIGURE 1

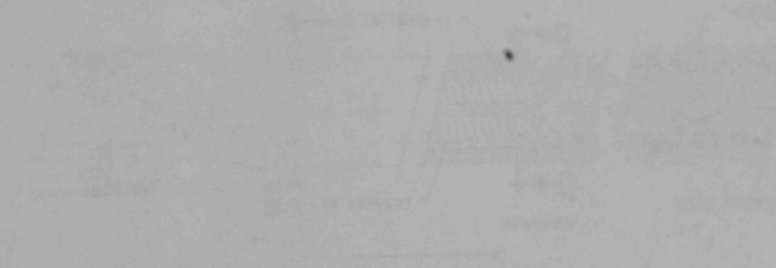


FIGURE 2

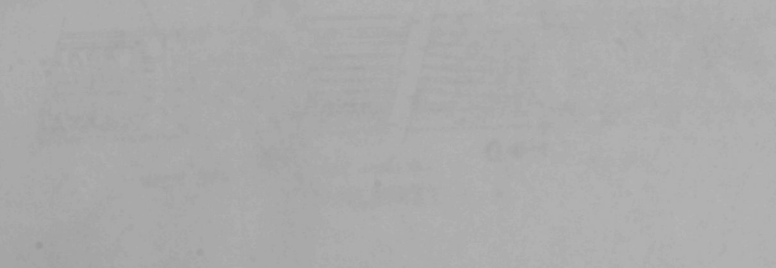


FIGURE 3

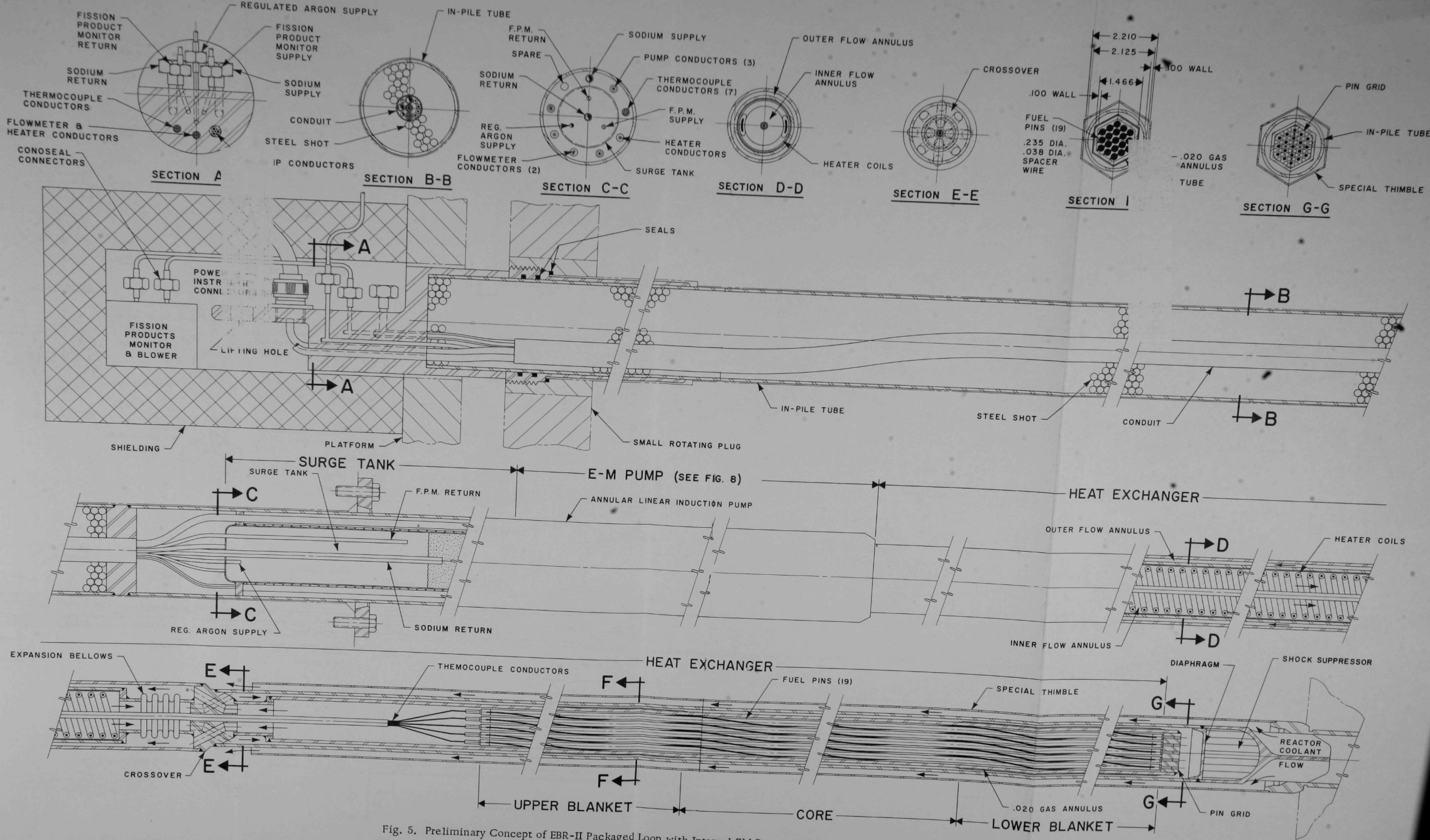


Fig. 5. Preliminary Concept of EBR-II Packaged Loop with Internal EM Pump and Concentric-tube Heat Exchanger

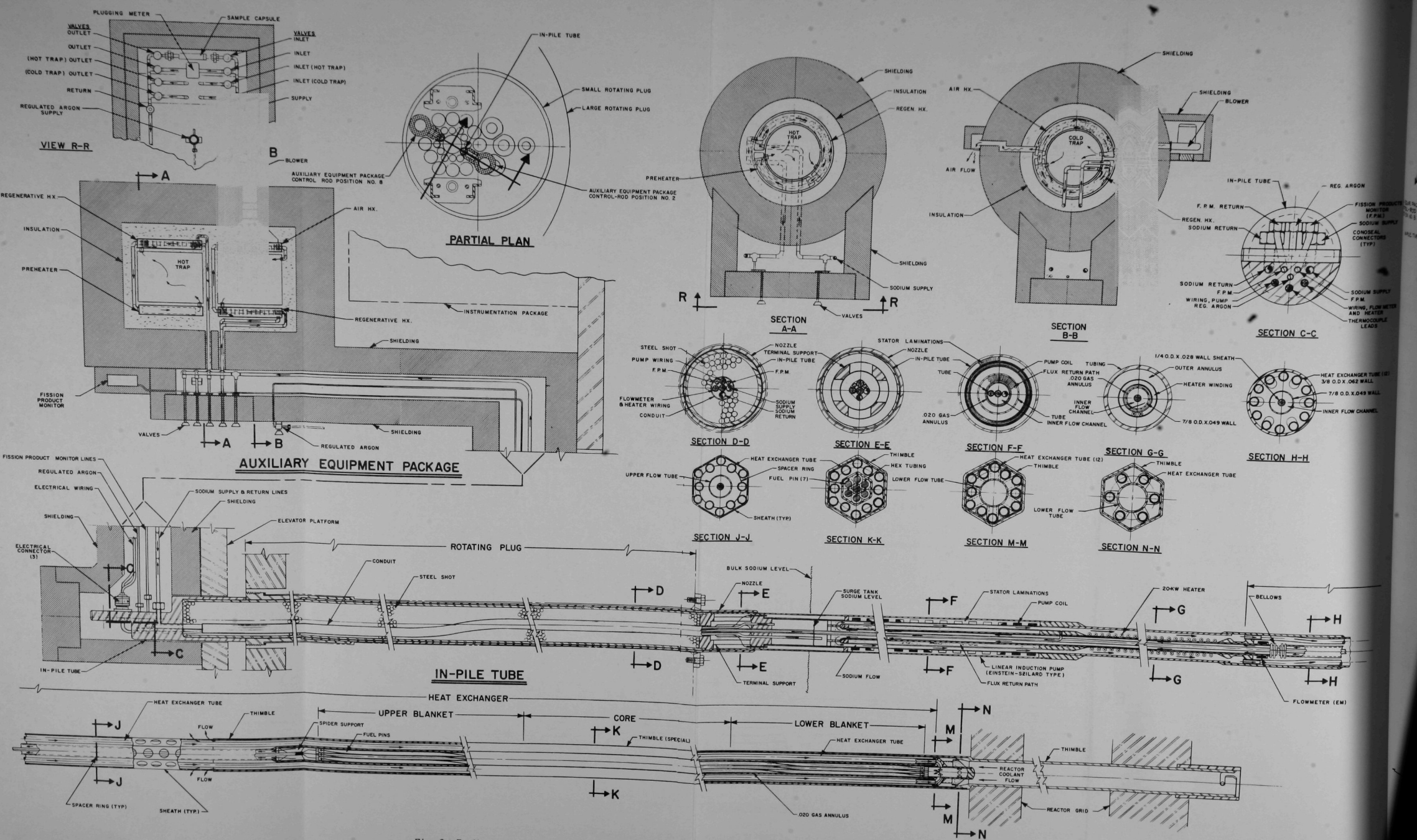


Fig. 6. Preliminary Concept of EBR-II Packaged Loop with Internal EM Pump and Multitube Heat Exchanger

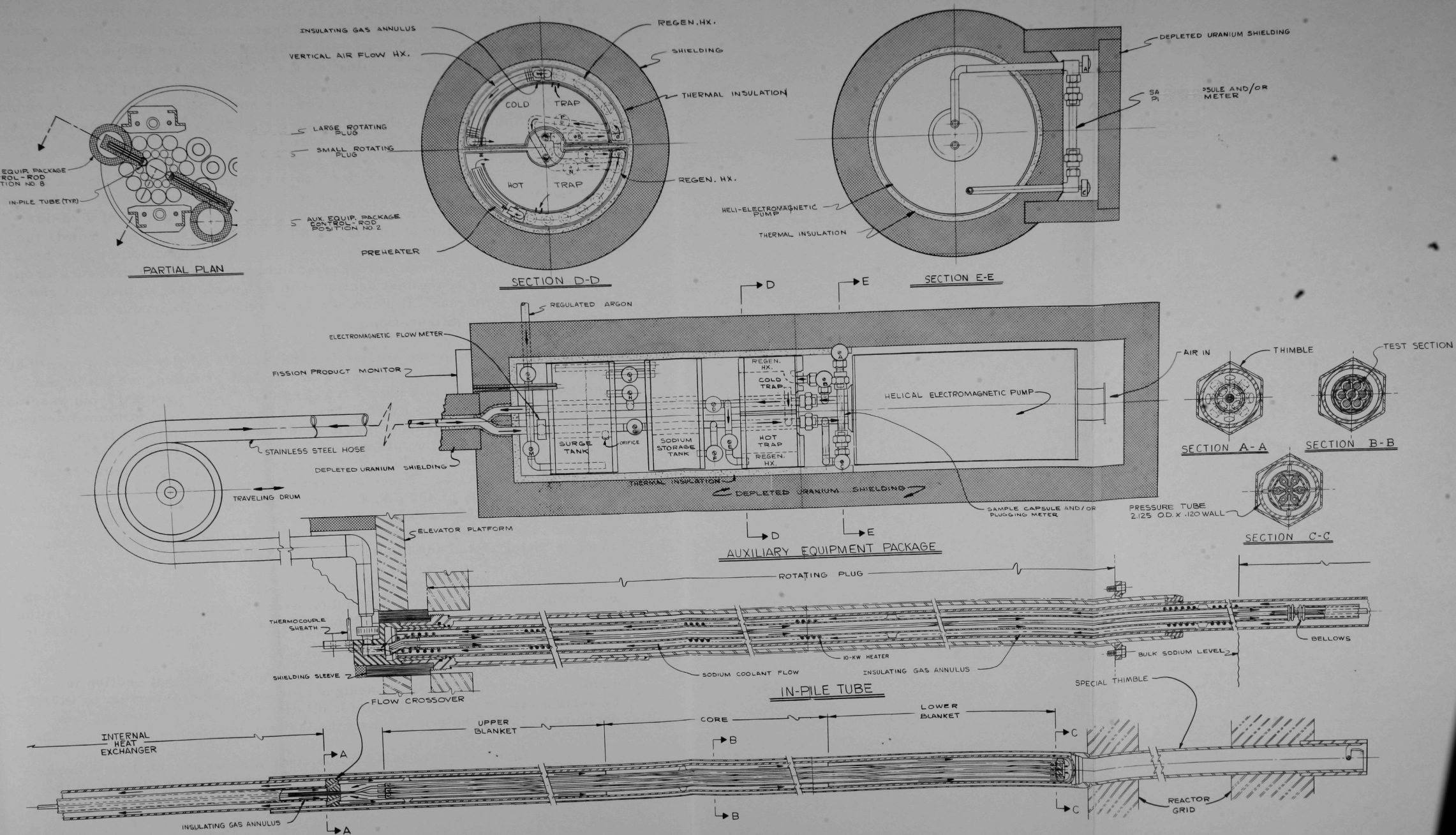


Fig. 7. Preliminary Concept of EBR-II Packaged Loop with External Pump

and 50 in. long. For illustration, the figures show subassemblies with 7 and 19 fuel pins. However, other fuel configurations may be used as required by the experimenter. Loop sodium is delivered to the bottom of the test section through an other annulus (as shown in Fig. 5) or through outer heat-exchanger tubes (as shown in Fig. 6). Flow is upward past the fuel pins. The small space between the loop-containment boundary and the surrounding special thimble allows a flow of reactor primary coolant for the removal of heat from the loop downcomer section.

The test cross section can conveniently accommodate a seven-pin assembly with 0.36-in.-dia pins or a 19-pin assembly with 0.235-in.-dia pins having 0.038-in.-dia spacer wires. A 19-pin assembly of 0.25-in. diameter with an interpin spacing of 0.060 in. is feasible, but crowds the space available. The higher flowrates and smaller downcomer flow area with this array will result in higher sodium velocities and pressure drops in the downcomer of the test section. An EM pump larger than that shown in Fig. 8, or a mechanical pump, would be required to produce the 70-gpm flowrate for this configuration.

Heat-transfer studies (see Section VI.J below) have shown that more than 535 kW can be rejected from the loop at the higher outlet temperatures and higher flowrates. A rating of 535 kW is for a 19-pin assembly with a linear power of 25 kW/ft. This is considered the minimum linear power at which mixed-carbide fuels would be economical in a large LMFBFR. The design flowrate in Ref. 1 was 35 gpm. Higher rates of 50 and 70 gpm have been considered for decreasing the differential temperature across the test section. Based on a core height of 13.5 in. and a heat-rejection capability of 400 kW, a 19-pin assembly could have a power density of about 20 kW/ft for each pin. Power densities of from 35 to 40 kW/ft for each pin in a seven-pin subassembly could be attained, depending on the fuel material being tested and the neutron flux available.

The fueled region of the test section is insulated from the loop downcomer by a thin inert-gas annulus, except for the 13.5-in. length in the reactor core. This section is left uninsulated for dissipating decay heat in case of a loss of loop flow.

Although the design exit temperature of the test section is 1200°F, a test section could be designed for exit temperatures up to 1475°F by using a bypass of inlet sodium for dilution to reduce the coolant temperature to 1200°F before it leaves the test section.

An out-of-core test section could also be provided for comparison testing in a low-radiation field above the upper axial reflector or the reactor-vessel cover.

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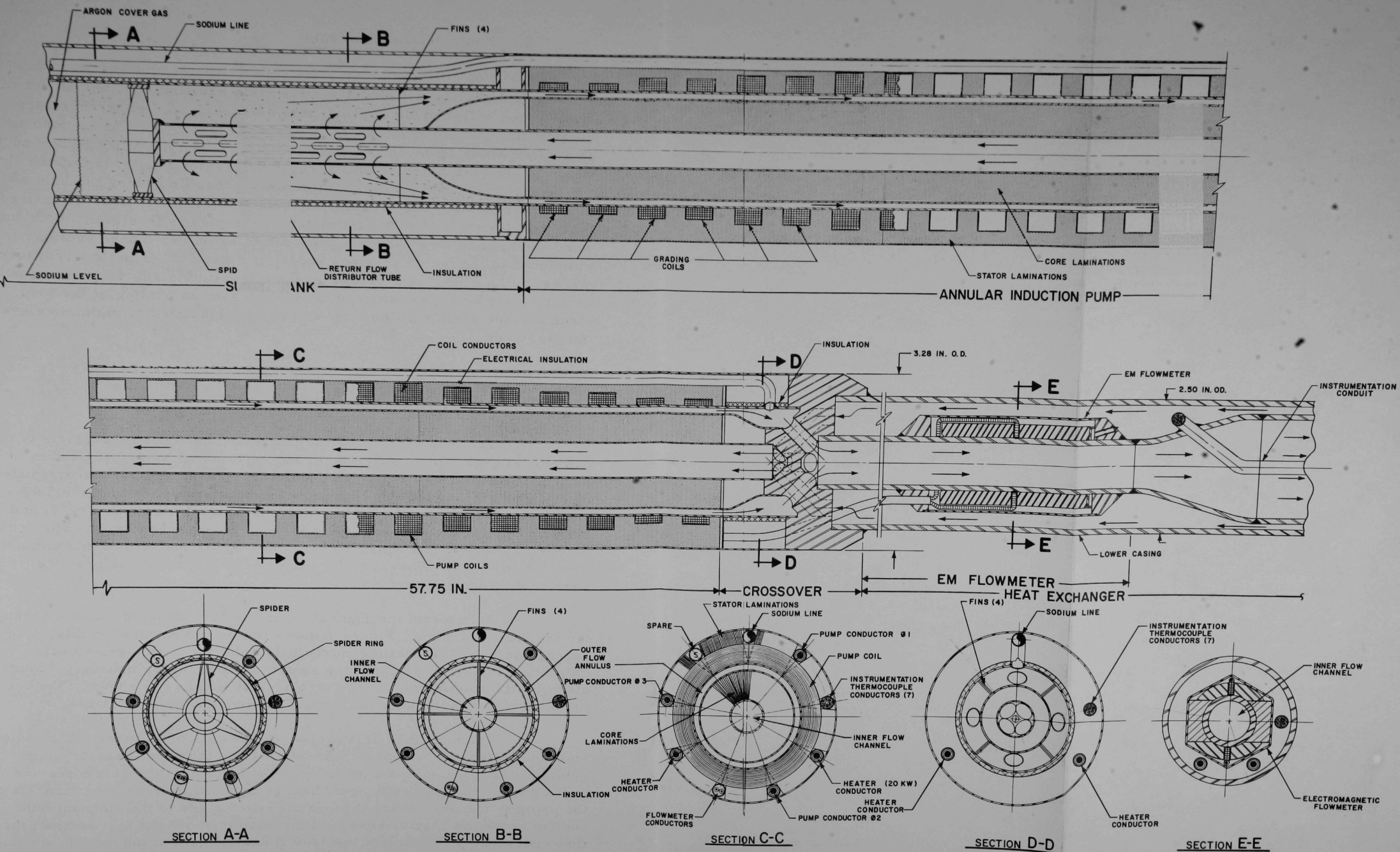


Fig. 8. Einstein-Szilard Reversed-flow Annular Induction Pump for EBR-II Packaged Loop

## 2. Nonfueled Experiments

For nonfueled materials or sodium-technology experiments, an in-core test section will provide a fast-flux environment, and an out-of-core test section will provide similar conditions such as high temperature and an activated sodium environment without the fast flux. The in-core section is a space about  $2\frac{1}{8}$  in. in OD and 50 in. long. The out-of-core test section, which is of equal diameter and of equal or greater length, is located above the reactor core in the outlet plenum of the reactor and/or the bulk sodium above the reactor. The experimenter can use these sections as desired. Since the direction of flow past material samples is irrelevant, both downward and upward flow channels can contain samples. These sections can be simpler than the fueled versions shown in Figs. 5-7. The flow crossover pieces, bellows, and heat exchangers can be eliminated. Since there will be very little heat generation (mainly from gamma heating of the loop structure and samples), the tube will be well insulated to maintain elevated temperatures.

### C. Loop System

#### 1. Internal-pump Concept

The loop system for the internal-pump concept consists of the pump, heat exchangers, surge tank, and containment tubing, as shown in Fig. 5. A preliminary design has been made for an annular linear-induction pump (Einstein-Szilard type) for high-temperature operation (see Section VI.A and Appendix A). A layout of the pump is shown in Fig. 8, and the pump design criteria are listed in Table X (in Appendix A). The control-rod-drive nozzle in the small rotating plug limits the size of this pump to about  $3\frac{1}{4}$  in. in OD. However, the pump can be as long as is practical to develop the required performance.

An internal mechanical pump has been considered as a backup for the induction pump. Figure 9 shows two arrangements for this. One arrangement uses the canned-motor principle; the other uses a sealed magnetic coupling to maintain the system high-pressure boundary. Multistage centrifugal and multistage turbine-type pumps are shown. The drive motor is a small-diameter unit, which is commercially available.

The multitube heat exchanger consists of 12 tubes, which extend from the bottom of the reactor-vessel cover to the upper reactor-support grid; heat is rejected to the primary coolant flowing through the reactor. In the concentric-tube exchanger, heat is rejected from an outer annulus to the primary coolant flowing through the reactor and to the stagnant bulk sodium above the reactor vessel. Immediately above the test section, the flow crossover piece diverts the 1200°F loop coolant to the outside annulus

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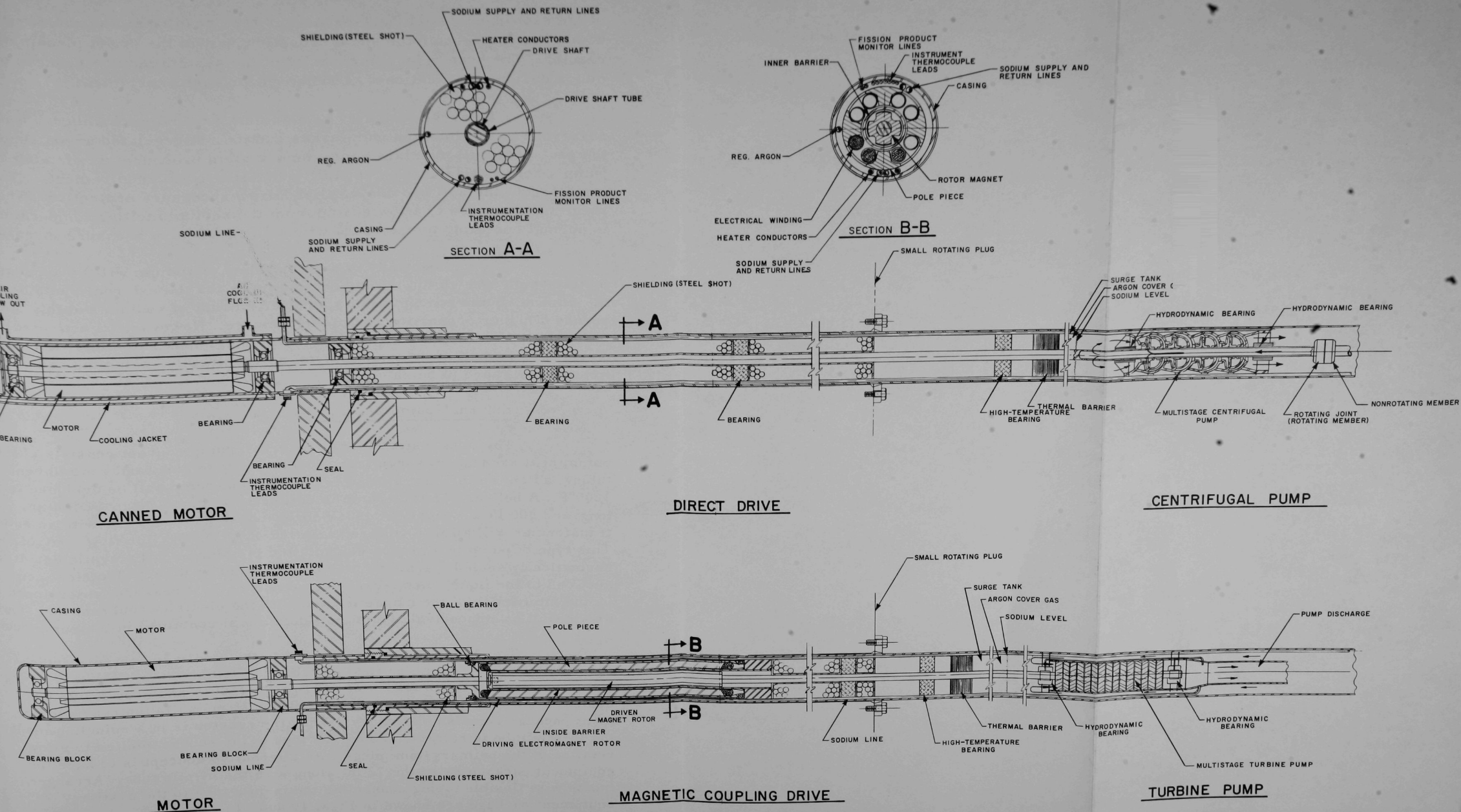


Fig. 9. Preliminary Concepts for Mechanical Pump Design for EBR-II Packaged Loop

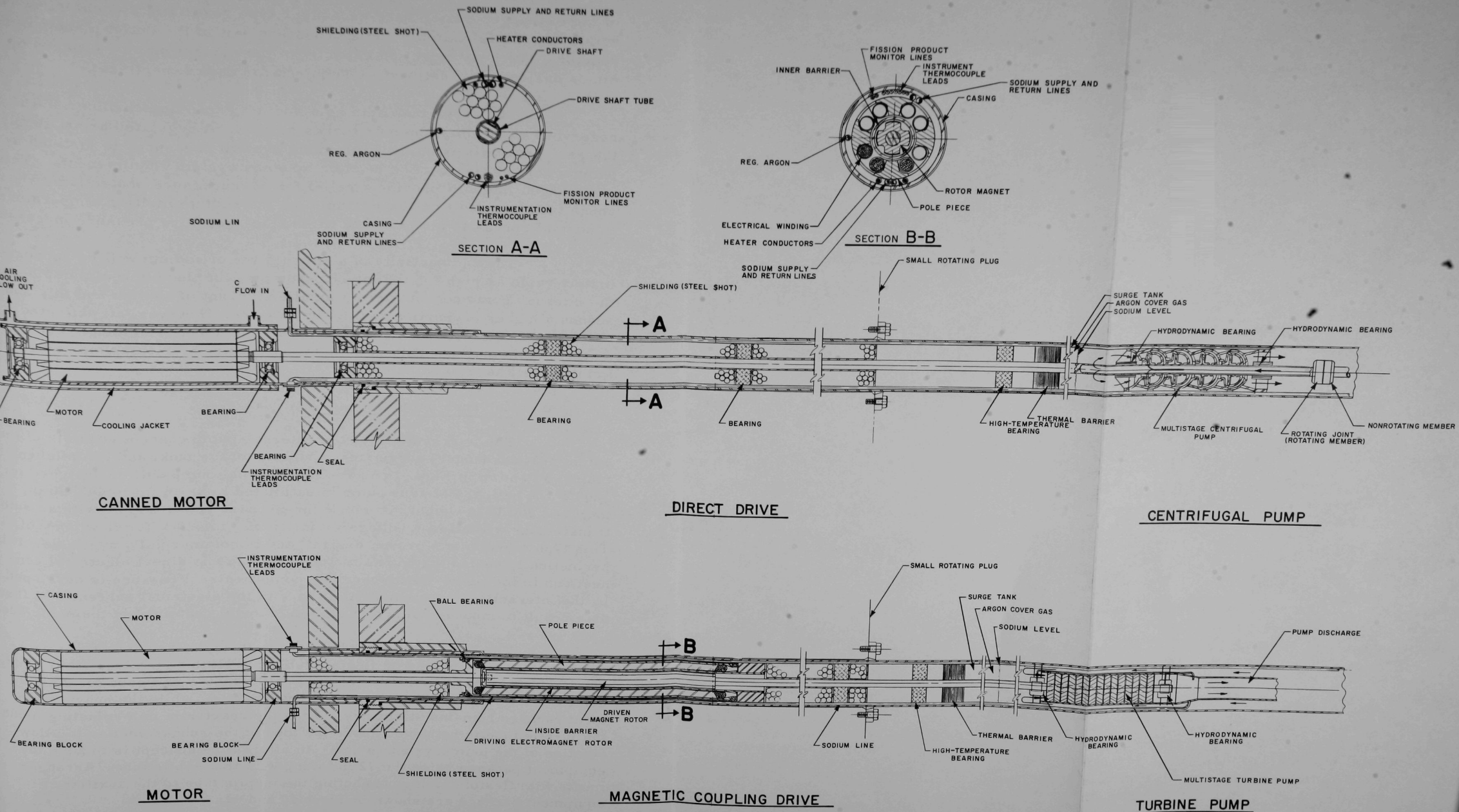


Fig. 9. Preliminary Concepts for Mechanical Pump Design for EBR-II Package Loop

of the exchanger. Each heat-exchanger type results in a counterflow unit within the special thimble and a crossflow unit in the outlet plenum of the reactor. The heat-rejection capacity of the system may be changed by minor design modifications, as required for each experiment.

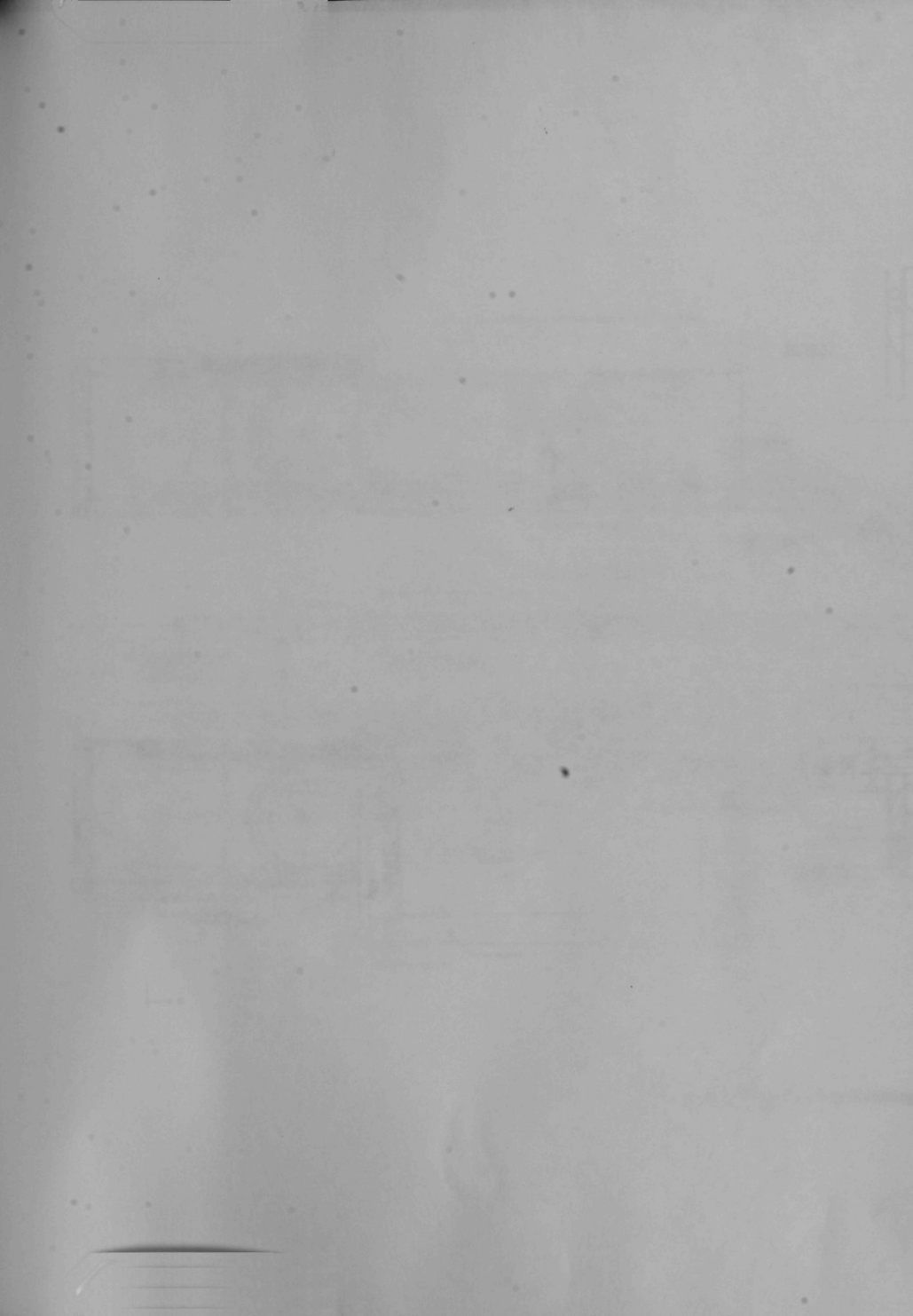
The surge tank acts as a sodium expansion chamber and is a reservoir from which the pump takes suction. A regulated argon supply to the gas space will maintain a minimum system pressure, as dictated by the pump's net positive suction head or by other system requirements. This same argon line will provide a relief for overpressure protection. The surge tank is sized to allow heatup from the sodium melting temperature to normal operating temperature without an excessive pressure increase.

The radioactivity of a unit volume of sodium will be approximately twice as high for the internal pump as for the external pump because the internal-pump concept uses a smaller amount of sodium and this sodium spends a higher proportion of time in the core. However, all activated sodium will normally remain within the primary tank shielding. For special tests, only a small quantity of sodium, as required for sampling and purification, will have to be shielded.

## 2. External-pump Concept

The loop system for the external-pump concept consists of the pump, heat exchanger, surge tank, sodium storage tank, and containment tubing, as shown in Figs. 4 and 7. The external pump will be designed for 1200°F. A helical EM-type pump is estimated to be 16 in. in OD, 30 in. long, and 600 lb in weight. A 440-V three-phase power supply with an auto-transformer will allow a voltage variation of 0 to 560 V for flow control. This type of pump is a proven design<sup>14</sup> and is commercially available. It is hermetically sealed, having no moving parts and no direct electrical connections to the liquid-metal-carrying components. Pressure is developed by the interaction of the magnetic field with the electrical current that flows as a result of induced voltage in the liquid metal contained in the pump duct.

The in-pile tube is connected to the auxiliary equipment package by either fixed piping or flexible tubing. Flexible stainless steel tubing (available for a rated pressure of 2400 psi at 1500°F and good for 25,000 cycles) allows the in-pile tube to be elevated 72 in. while the heavy equipment package remains stationary. When the reactor is in a refueling mode, the in-pile tube is retracted above the tops of the subassemblies to allow rotation of the primary-tank plugs. An alternative concept is to have the equipment package raise the 72 in. along with the in-pile tube. Arrangements with flexible tubing connecting the in-pile tube to the auxiliary equipment package are shown in Figs. 10 and 11.



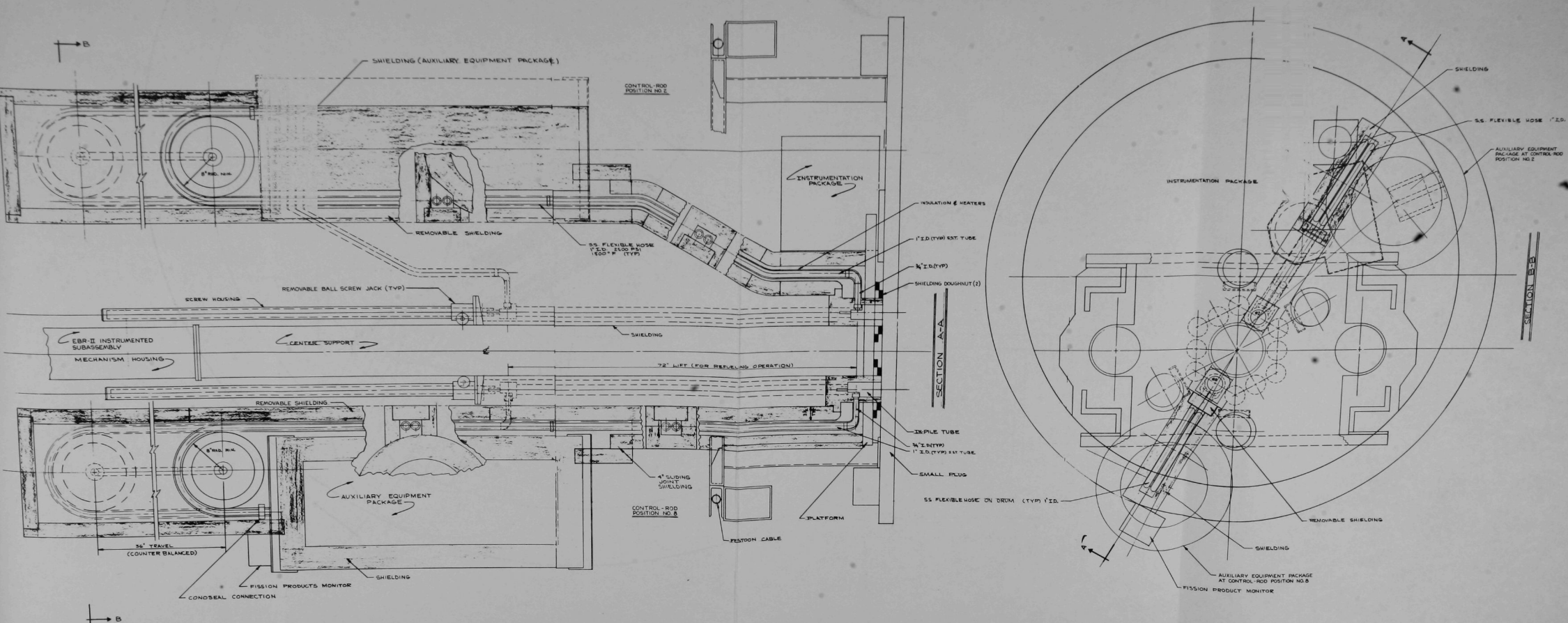


Fig. 10. EBR-II Packaged-loop Shielding and Equipment Package: Upper Connection Installation

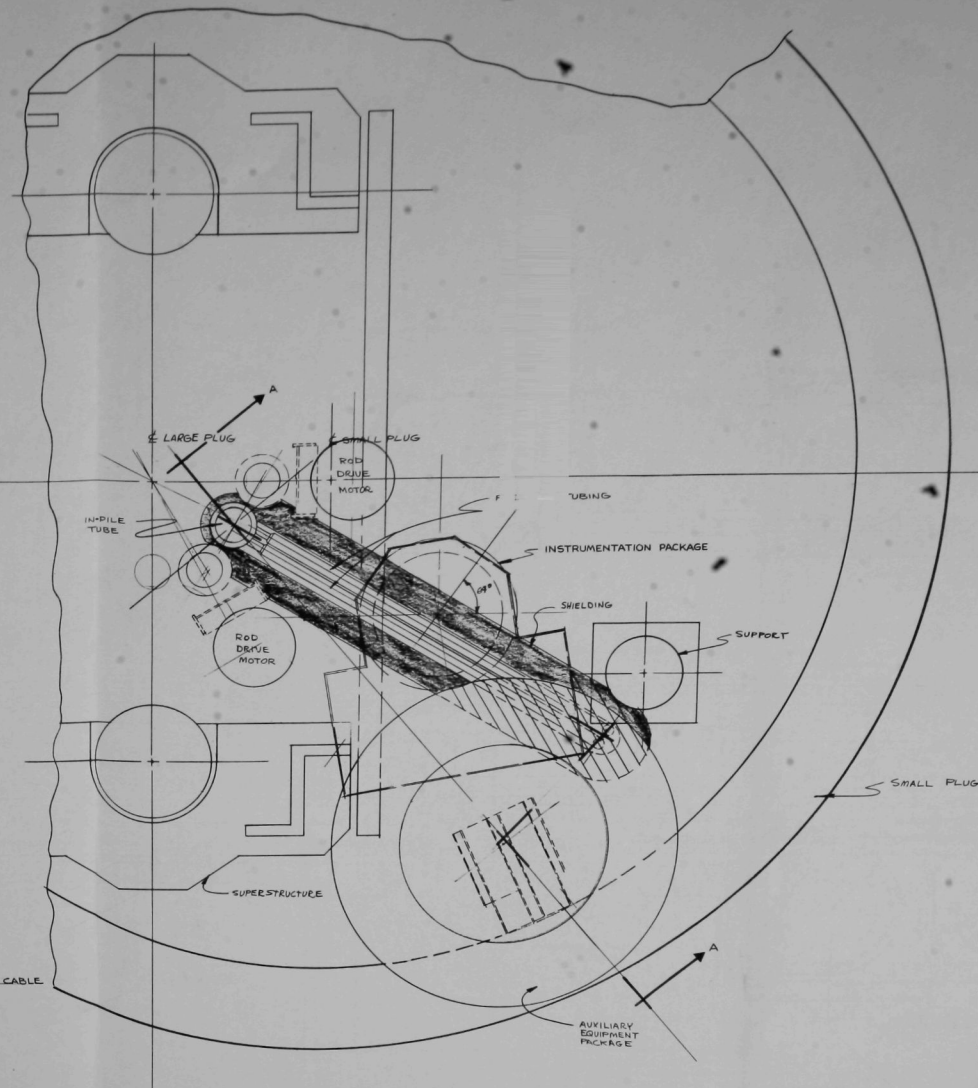
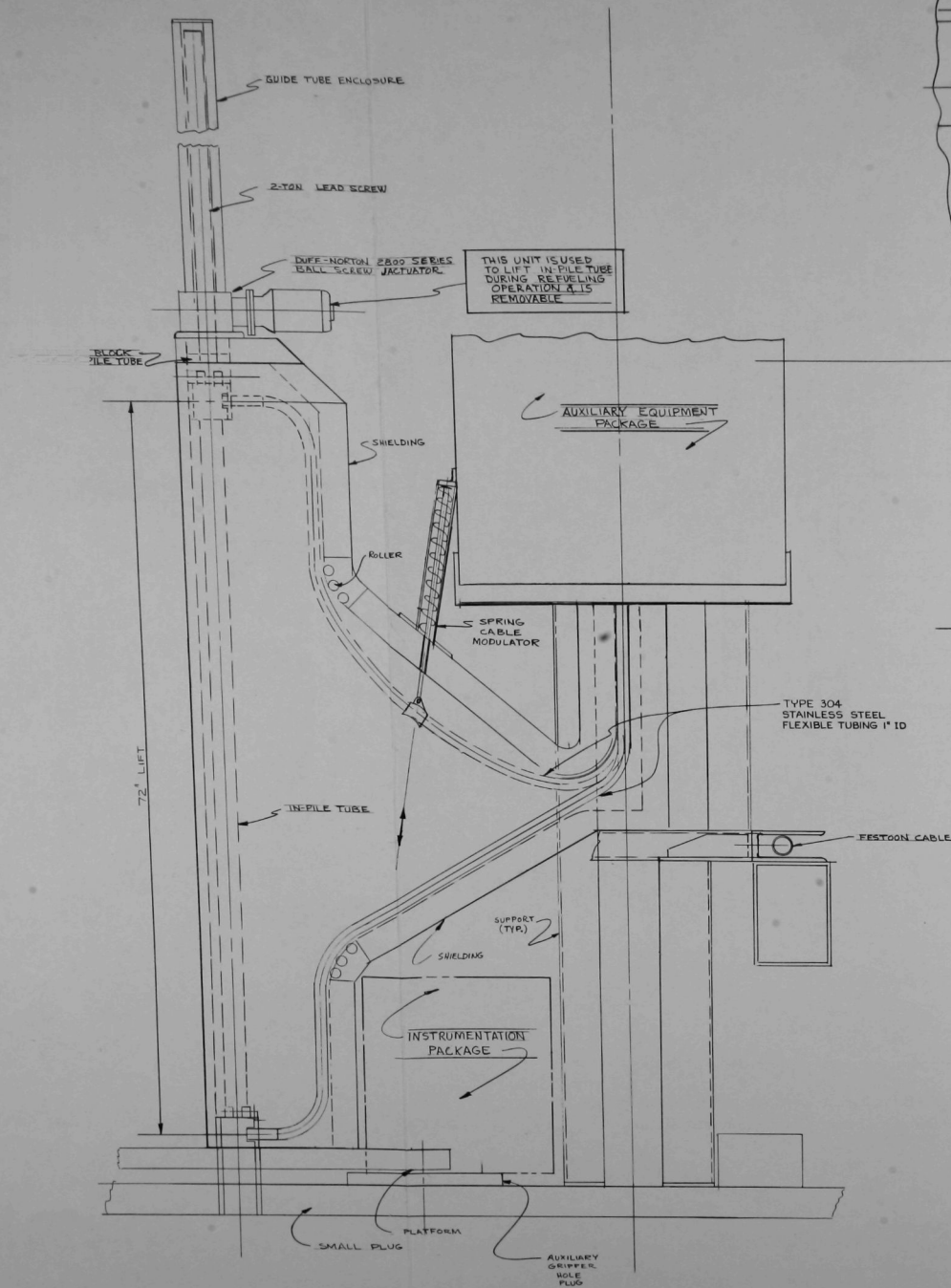


Fig. 11  
EBR-II Packaged-loop Shielding and Equipment  
Package: Lower Connection Installation

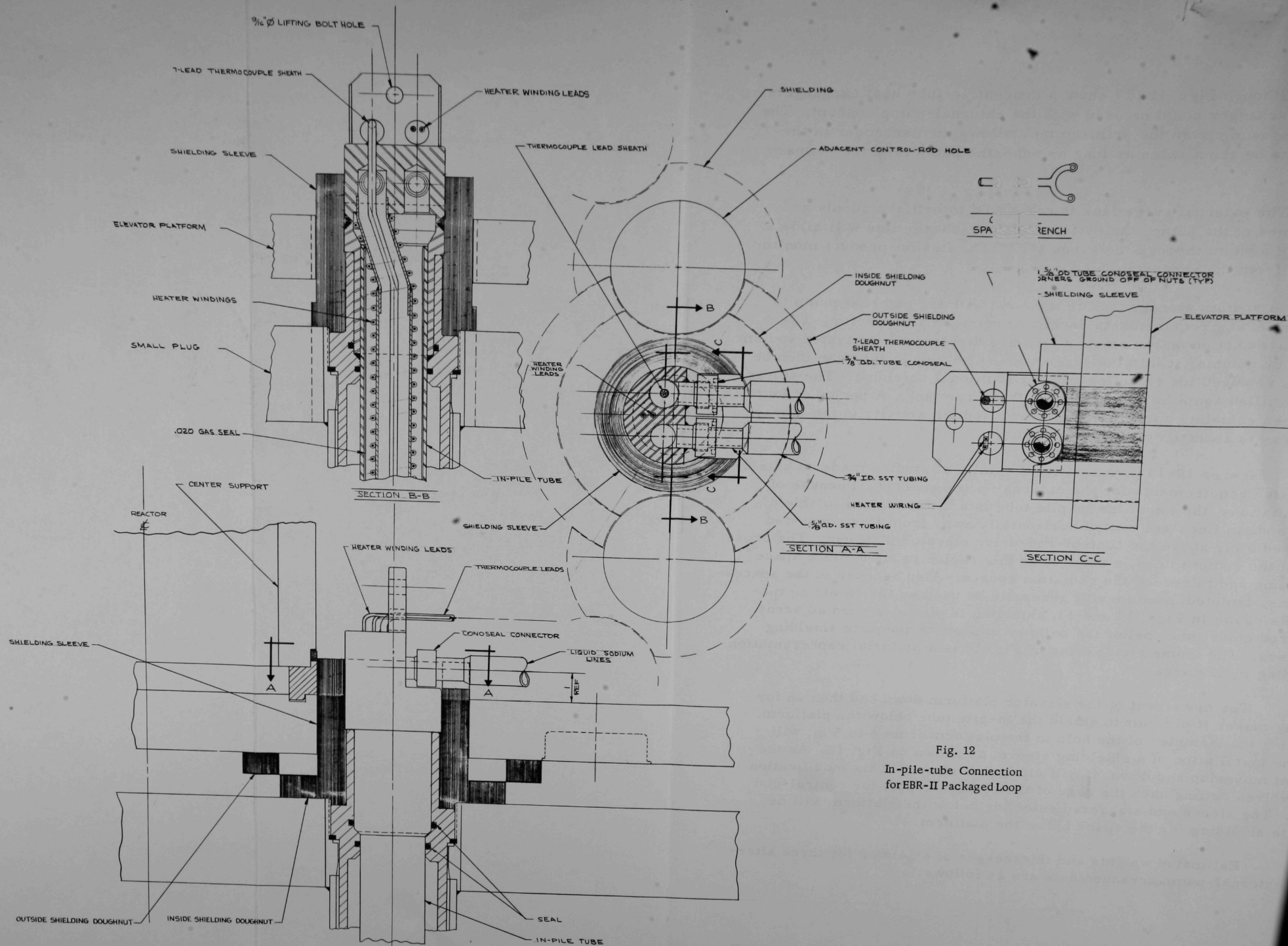


Fig. 12  
 In-pile-tube Connection  
 for EBR-II Packaged Loop



Although Figs. 4 and 7 show a concentric-tube heat exchanger, a multitube exchanger could be used with the external-pump concept. The removal of the pump from the primary tank allows the exchanger in the bulk sodium to be about twice as long as when the pump is in the primary tank.

The external surge tank is connected to both the supply and discharge lines of the pump. An orifice in the discharge line will allow a small flow of sodium through the surge tank. The fission-product monitor will sample from the surge-tank vapor space.

A sodium storage tank is included with the external pump to help meet shielding requirements during refueling of the reactor. The pump can take suction on the storage tank and displace the activated sodium in the loop by pumping it into the surge tank. The in-pile tube can then be elevated as required for refueling without additional shielding. The storage tank can be filled again by draining from the surge tank. A period of four days between refueling operations will allow sodium activity in the storage tank to decay to about 1% of its initial value.

Figures 10-14 give additional details of studies made on shielding and space requirements for the external-pump concept. Because of space limitations, the top of the in-pile tube is a difficult area to shield. The loop sodium lines are routed horizontally away from this congested area immediately above the control-rod-drive elevator platform. The cross section of the sodium lines is reduced in this region to allow more space for shielding and to reduce the radiation source. Also because of the space limitations, depleted uranium will generally be used as the shielding material. As shown in Figs. 13 and 14, shielding is placed around adjacent control-rod guide tubes below the bellows to provide adequate shielding thicknesses. The isometric drawing of Fig. 14 is a pictorial representation of shielding in this area.

The movement of the elevator platform down and then up for refueling makes it difficult to shield the in-pile tube below the platform. Increasing the diameter of the hole in the platform from 4 to 5 in. will allow the installation of a shielding sleeve, as shown on Fig. 12. As the platform moves up and down, it will clear this sleeve. This modification also involves cutting back the base of the center support for control-rod drives. The sleeve and staggered shielding below the platform will be adequate shielding for the space below the platform.

Estimated weights and thicknesses of shielding for three alternative external-pump arrangements are as follows:



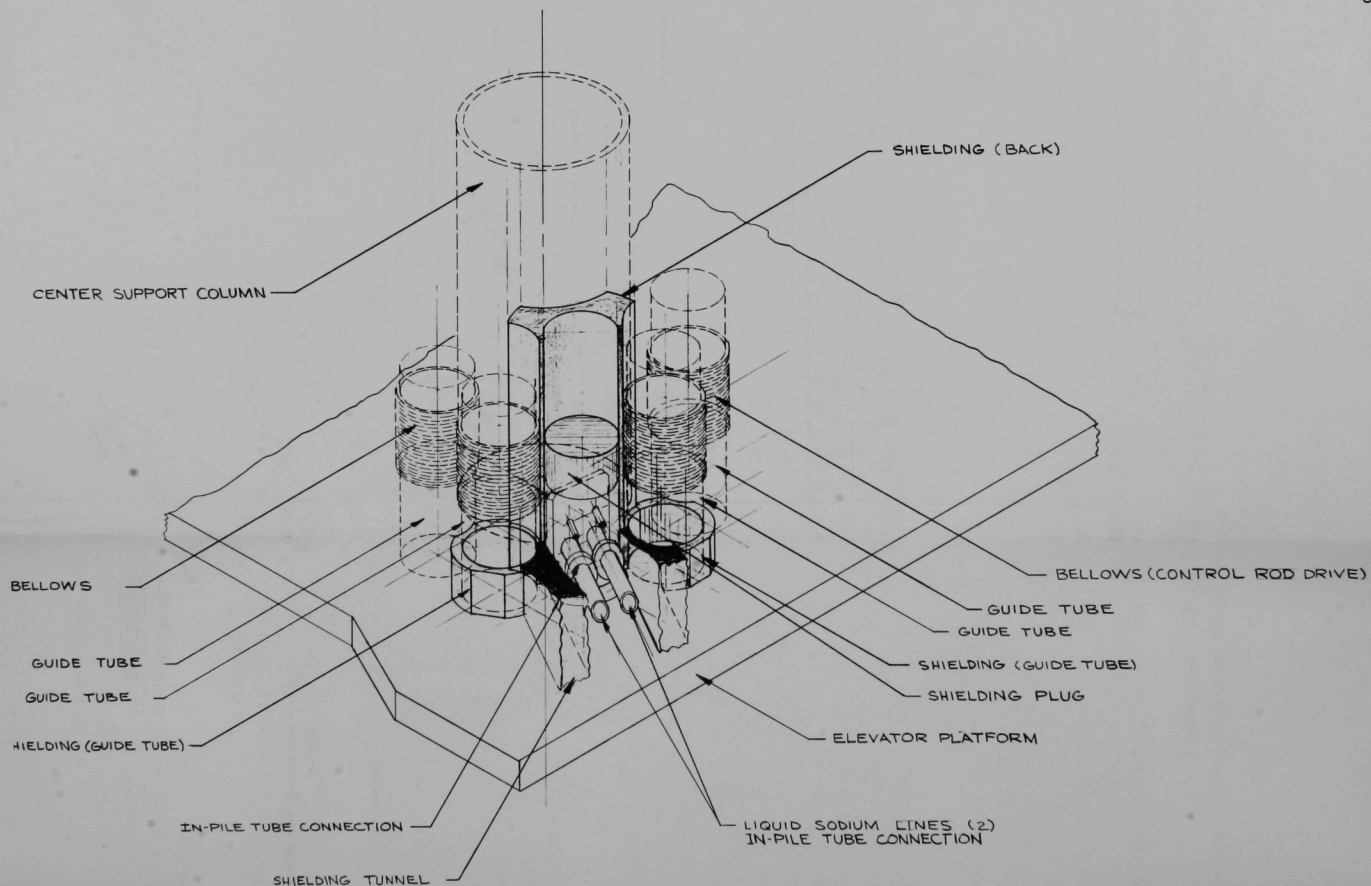


Fig. 14. EBR-II Packaged-loop Shielding: Isometric View

<u>Location</u>	<u>Shielding Thickness, in.</u>	<u>Shielding Weight, tons</u>
Equipment package	5	8.5
Sodium lines:		
1. Fig. 10 (flexible tubing with top connection)	3.5	6.5
2. Fig. 11 (flexible tubing with bottom connection)	3.5	6.0
3. Fixed piping with bottom connection	3.5	1.5

The installation of two packaged loops with external pumps would increase the load on the small-rotating-plug bearings by about 25 to 35%, depending on the type of installation considered. Most of this weight would be shielding. A preliminary study shows that there would be no stress or deflection problems in the small- or large-plug bearings or other load-bearing components.

### 3. Auxiliary Components

The one auxiliary component required for normal operation is the fission-product monitor. This would be similar in principle to the EBR-II reactor cover-gas monitor. A stream of loop cover gas would be monitored by a detection system consisting of a scintillator, a photomultiplier, and a single-channel analyzer, which is discriminated for gross fission-product isotopes.

Components that require special testing are hot and cold traps, a plugging meter, and the sodium sampling station.

The cold trap operates in conjunction with an air-cooled heat exchanger and a regenerative heat exchanger. Since the maximum sodium volume in the system, including the surge tank, is about 8 gal, a flow of 10 gph through the 1-gal-capacity cold trap is more than sufficient for coolant purity control.

The hot trap is included for experiments requiring high coolant purity. The sodium that goes to the hot trap is heated in a preheater with high-density electric heaters. Flow is established through both cold and hot traps with EM-type flowmeters and throttling valves. Valves are operated with reach rods that penetrate the radiation shielding.

The sodium-sampling station is partially shielded from the other equipment and is accessible by removing a shielding cover from the outside. A sample of sodium can be removed from the operating system for purity analysis. Impurities can also be added to the system at the sampling station to establish known concentrations for test purposes, and operation can be continued with or without the sodium-purification system in operation. The plugging meter and other experimental equipment will also be located at this station.

#### D. Instrumentation

Sodium temperature is controlled by means of electric heaters and by varying the loop flowrate. Electric heaters are located in the in-pile tube above the bulk-sodium level and in the external-pump equipment package. In the internal-pump arrangement, the electric heaters are located in the in-pile tube below the pump. The flowrate is varied by changing the power-supply voltage to the EM pump or the frequency to the ac motor driving the mechanical pump.

Thermocouples for operational control are located at the inlet and outlet of the test section. Coolant temperature control can be based on either of these thermocouples, depending on the experiment being irradiated. Additional thermocouples may be located in the pump, heat-exchanger sections, and flowmeter. Thermocouples can be located to measure fuel temperatures or at any point of interest to the experimenter. The instrumentation conduits can be larger than shown in Figs. 5-7 to accommodate more instrumentation leads as required by the experimenter. Although not shown in the figures, pressure instrumentation can be routed to any part of the loop test section.

#### E. Loop-handling Equipment

Equipment for removing a radioactive in-pile tube of the packaged loop is shown in Fig. 15. The equipment consists of a coffin and containment tubes, which will fit both below and above the coffin when it is in position to receive the experimental test section. During loop handling, the coffin will rest on the structure used for lifting the reactor-vessel cover. The coffin will be located directly over the in-pile tube after a lower containment tube has been placed in position to provide a sealed inert space between the coffin and the top of the small rotating plug.

The upper containment tube is placed on top of the coffin and is steadied by the crane in this position. The upper containment tube contains two drives for elevating the in-pile tube: a normal ac drive, and an emergency dc drive with battery-pack power supply. It also contains a shearing mechanism to cut and seal the in-pile tube, thus leaving the radioactively

SECTION D-D

Vertical distance of bearing structure to ground level

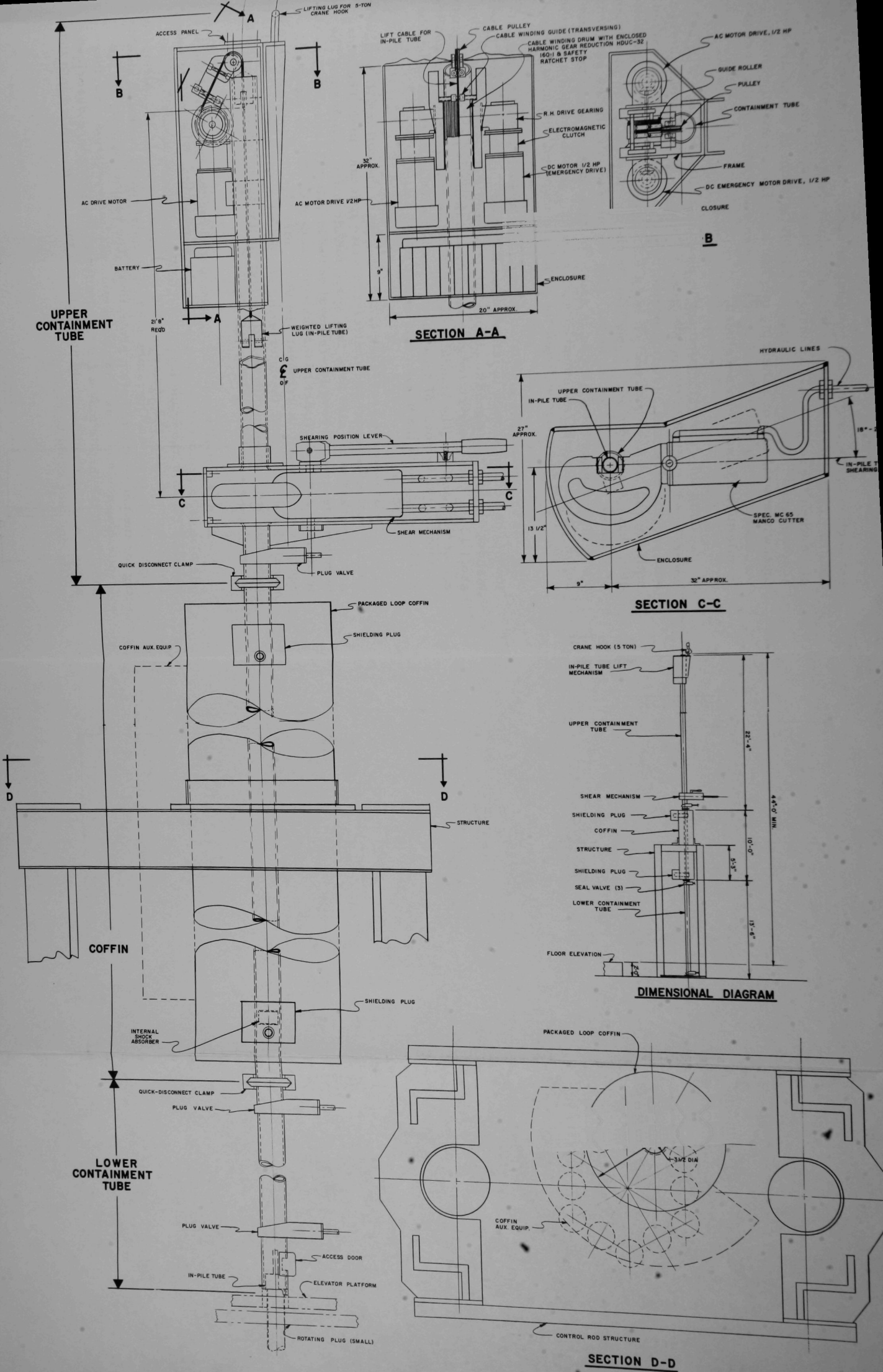


Fig. 15. Preliminary Concept of Handling Equipment for EBR-II Package Loop

hot section in the coffin. The handling of preirradiated fuel will require handling of the 30-ft length of coffin and upper containment tube in one piece when the in-pile tube is installed. Therefore, removal might be accomplished in the same manner as the installation of the preirradiated fuel, without the use of a shearing mechanism. The means of removing a test section from the coffin after it has been severed from the rest of the in-pile tube has not been developed.

With the removal equipment in place, a cable and hook will be lowered and attached to the in-pile tube. Distance and movable shielding will provide personnel radiation protection as the in-pile tube is raised, until the radioactively hot section is safely inside the coffin and the bottom gate shut. A second mode of operation being considered is the placement of portable shielding around the test section instead of near the personnel while the irradiated test section is moved between the primary tank and the coffin. A possible arrangement of such shielding is shown in Fig. 16. (See Section VI.I.2 for a discussion of the problems involved with this concept.)

When the in-pile tube is at the top of its travel, it will be cut, allowing the bottom section to fall onto the lower coffin gate. On this gate is a shock absorber designed to absorb the impact of the test section falling about 12 in. The upper coffin gate is shut, and then the removal equipment can be dismantled. The specially designed coffin will have its own cooling system for the removal of decay heat. The top section of the in-pile tube will be removed in the upper containment tube.

To provide the necessary crane clearance, the coffin will be lowered into the space just above the control-rod-drive mechanisms. The instrumented subassembly also uses this space for elevating equipment. The technique described above for removal of the in-pile tube may require the temporary removal of part of the instrumented-subassembly elevating equipment. In-pile tube removal will be an infrequent operation, and minor modifications will make the interfering equipment easily removable. Therefore, this problem should have little effect on the operation of either system and negligible effect on the plant factor.

An alternative method being considered for the removal of the in-pile tube is to remove it in one piece to the storage pit. The in-pile tube would be lifted by the crane into a containment tube, which would be shielded to attenuate the radiation field by a factor of about 10. All operation would be remote. Operators would use distance and operator shielding for the necessary radiation protection. The remaining work on the in-pile tube could be done without affecting reactor operation. Further handling of the in-pile tube would be similar to the procedure described above. After the radioactively hot section of the tube was pulled into the coffin, the tube could be cut.

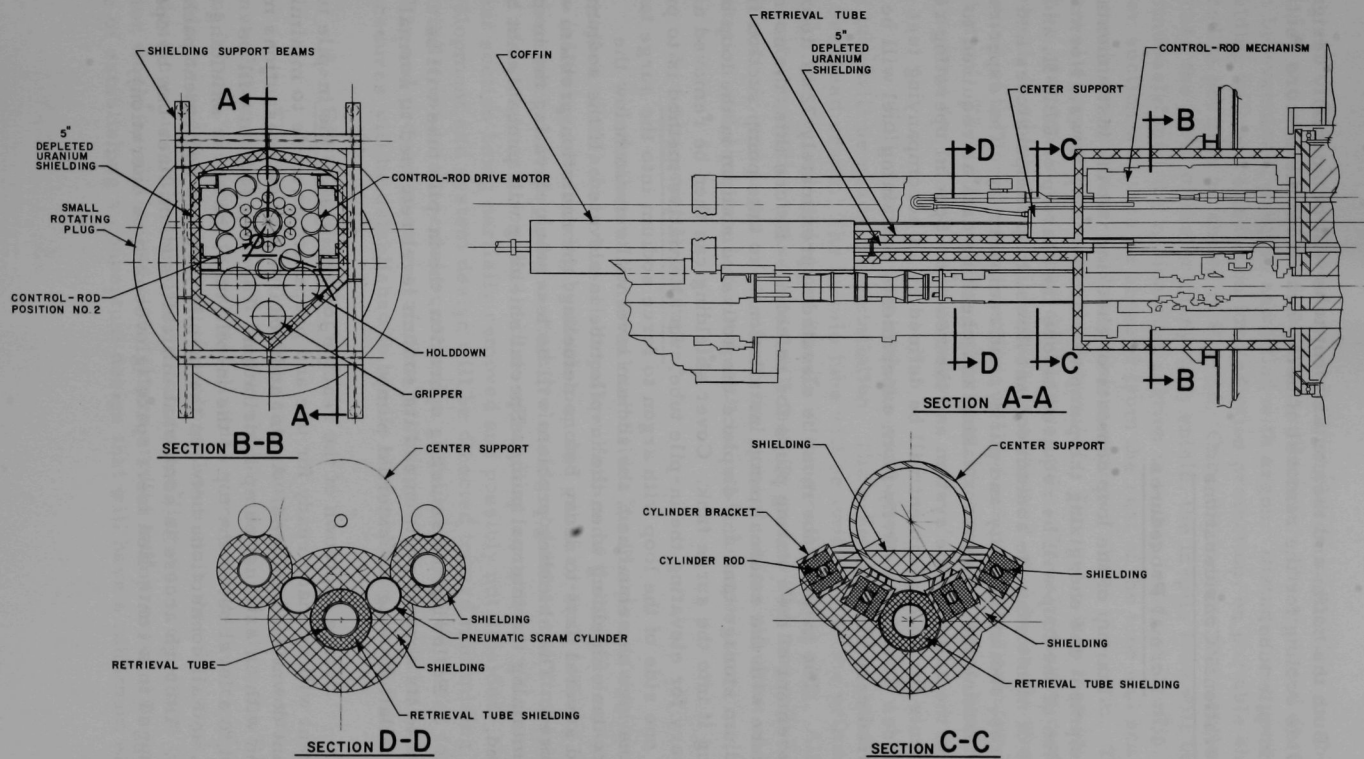


Fig. 16. Shielding for Removal or Insertion of EBR-II Packaged Loop

Both the coffin and the containment tube will be specially designed to provide cooling for the removal of decay heat from the in-core section of the in-pile tube.

## F. Operational Considerations

### 1. Normal Procedures

Startup of the loop consists of checking out the instrumentation and equipment and energizing the pump to establish loop flows. Normal operation of the loop will be required before the startup of EBR-II, and will consist of monitoring the sodium coolant flows, loop temperatures and pressures, sodium quantity and purity, and radioactivity. The experimenter will determine the interval for taking and checking data as required to maintain the safety of the system and the reactor. Special operating requirements for an experiment will be defined by an accompanying test procedure. The radioactivity from experiments containing fuel will be monitored very closely.

The in-pile tube must be elevated approximately 72 in. to permit movement of the rotating plugs for refueling. Before this, the normal procedure with the external pump installation is to take pump suction from the sodium storage tank and displace the activated sodium in the loop by pumping it into the surge tank. Cover shielding can then be removed as necessary for elevating the in-pile tube. An alternative method is to pressurize one side of the loop with argon to force sodium into the surge tank. When the legs are equalized, the sodium level will remain below the primary-tank shielding when the in-pile tube is elevated. If the sodium has had several days to decay before refueling, the radiation problem will be reduced. The shielding problem will be less when elevating the in-pile tube containing the internal pump. Special shielding will probably not be required.

Following the refueling operation, the in-pile tube will be lowered to its normal position and its sodium level returned to normal in the installation using the external pump.

Before shutdown of the reactor for removal of the in-pile tube, equipment will be made ready for use and preparations made to minimize any plant downtime required. Again, additional work and precautions must be taken with an external pump installation. The loop sodium will be drained to a level below the top of the in-pile tube, the cover shielding removed, and all connections between the in-pile tube and equipment package broken. Although access is somewhat difficult, sodium lines can be opened and plugged in a controlled inert space (glovebox-type operation).

The removal equipment will be placed in position, as shown in Fig. 15, and attachment made to the in-pile tube. The enclosed system can then be evacuated, purged, and filled with argon. Distance and personnel shielding will be relied upon for radiation protection, or portable shielding placed in position as shown in Fig. 16. Once the in-core test section reaches the level of the primary-tank shield, the in-pile tube will be raised continuously until the radioactive section is safely inside the coffin. The lower section will then be sheared from the rest of the tube and sealed by the combination shear-seal mechanism mounted above the coffin. The removal equipment can then be dismantled. The upper in-pile-tube section will be removed in the upper containment tube. The alternative method of removal, in which the in-pile tube is not sheared, may be used. Further handling of the in-pile-tube sections will be independent of reactor-plant operation. The coffin will have provision for removing decay heat from the test section for as long as necessary.

Following the removal of a completed experiment, a new in-pile tube will be ready for installation. The assembly will be handled by the overhead crane. The in-pile tube will be lowered into the primary tank through the control-rod-drive nozzle in the small rotating plug. All piping and instrumentation connections will then be made. The loop will be filled with sodium, and argon vented from the system. After checkout of instrumentation and equipment, the loop will be ready for operation.

## 2. Abnormal Procedures

A number of abnormal conditions could arise during operation. Examples of these are loss of loop flow, failure of the in-pile containment tube (allowing intermixing between loop and primary-tank sodium), failure of the external loop-piping system (releasing sodium to the atmosphere), loss of instrumentation, and failure of handling equipment, which results in a test section being partially removed and possibly not shielded. Component development and system design will be directed toward minimizing the possibility of an abnormal occurrence. Equipment will be tested for reliability and proved in an out-of-core prototype loop. In addition, appropriate procedures will be established to handle such abnormal conditions.

## 3. Effect on Plant Factor

The installation, maintenance, normal operation, removal, and reinstallation of the packaged loop would have some effect on the EBR-II plant factor. If removals and installations were scheduled during planned shutdowns and the availability of personnel was such that this work could be done concurrently with other plant-controlling work, the effect on plant factor would be minimized. Any future studies will proceed with an objective of establishing a conceptual design that will have a minimum effect on

the reactor plant factor. Although the effect is difficult to estimate, loop operation might reduce plant factor by as much as 2-3%. This number is based on (a) the internal pump concept, (b) two removals and installations per year, (c) a normal number of reactor startups and shutdowns, and (d) a normal number of refuelings. Time required because of equipment malfunction has not been considered. The external pump arrangement might affect the plant factor by as much as 3-5%.

## VI. SUPPORTING RESEARCH AND DEVELOPMENT

Most of the out-of-pile testing and development of components would be accomplished in a proof-test loop. Individual components that would need to be developed and proven are discussed below.

### A. Internal Pumps

Preliminary estimates indicated that an annular linear-induction pump (Einstein-Szilard type) would need to be about 8 ft long if winding temperatures were maintained below 1000°F. Degradation of electrical insulation, increased coil resistivity, and the reduction in the ferromagnetic properties of the iron core at high temperatures are problems in the internal-pump application. A prototype pump would have to be developed and proven.

A preliminary design of an internal Einstein-Szilard type of annular linear-induction EM sodium pump has been completed. This design uses new high-temperature alloys, which minimize the above problems and greatly reduce the length of the pump. Pump-design information and calculations are included in Appendix A. A layout and pump-performance curves are shown in Figs. 8 and 17, respectively.

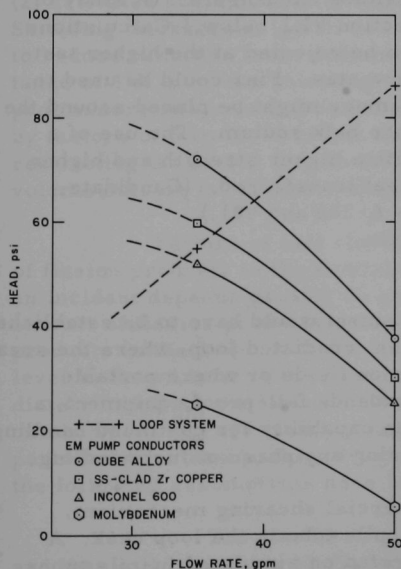


Fig. 17. Performance Curves for EM Pump and Packaged Loop

Figure 9 shows the layout of two internal mechanical pumps. The motor is located above the small rotating plug, and the extended shaft drives the pump, which is below the top shield plug. There is some experience<sup>15,16</sup> with internal mechanical sodium pumps to indicate they could be practical in the packaged loop. As design proceeded with one or more types, a prototype pump would have to be selected, built, and proof-tested.

### B. Seals and Connections

Low-pressure seals (to seal the in-pile tube to the rotating plug) and high-pressure sodium connections as shown in Fig. 12 should be proof-tested in a prototype installation. Although Conoseal sodium-piping connections have proved reliable<sup>17</sup>

under similar environmental conditions, this application should be tested before installation in the reactor. The internal-pump arrangement could use an expanding bellows similar to that used by the EBR-II instrumented subassembly for the low-pressure seal.

#### C. Hot- and Cold-trap Systems

Although sodium-purification methods are well standardized, prototypes of the miniature-sized hot and cold traps with their heat exchangers and preheater should be fabricated and tested.

#### D. Special Internal Loop Flowmeter

A permanent-magnet flowmeter of the type designed for the EBR-II instrumented subassembly<sup>18,19</sup> is required for the internal pump installation. The limited space and higher sodium temperature in the packaged loop would require performance and stability testing of a prototype.

#### E. Heat Exchangers

Testing should be conducted to verify calculations on heat-transfer capabilities of the concentric-tube or multitube exchangers. (A study of heat-rejection capability is included in Section VI.J below.) Calculations indicate that more than 535 kW of heat can be rejected at the higher test-section outlet temperatures and higher flowrates. Fins could be used to increase the heat-transfer area, and a chimney might be placed around the heat-exchanger section to induce flow in the bulk sodium. The use of a material in the heat-exchanger section with a higher strength and higher thermal conductivity could increase the heat transferred. (Candidate Carpenter alloys with higher strength are A-286 and 901.)

#### F. Loop-handling Equipment

The reliability of all handling equipment would have to be established through a test program. The removal of an irradiated loop, where the area has to be evacuated because of high radiation fields or where portable shielding (as shown in Fig. 16) is used, demands fail-proof equipment, a feasible means of equipment repair, or the capability for continued handling of the loop following equipment failure during any phase of fuel handling.

Figure 15 shows a standard commercial shearing mechanism, which could be used to cut and seal the in-pile tube in the loop cask. A shear would be designed, procured, and tested on simulated in-pile tubes. A redesign of the cutter shapes might be necessary to successfully seal the tubing. A means for handling the test section after it has been severed would have to be developed. The possible removal of the coffin and upper containment tube as one assembly would eliminate the need for the cutting mechanism.

## G. Flexible Hose Tests

If a concept with an external sodium pump and flexible tubing were chosen, the potential hazard involved in a failure in the flexible metal hose would require a proof-test program to establish the acceptability of commercially available flexible tubing.

## H. Sodium Technology

There would have to be adequate capability for determining the experimental sodium conditions. A large part of this capability is being developed as a part of EBR-II Task 18. A prototype loop with hot and cold traps, a sample capsule, a plugging meter, etc., could be used for out-of-pile development work in this area.

## I. Shielding Studies

### 1. Shielding during Loop Operation

Preliminary calculations have been made to estimate radiation fields near the external loop piping in the external pump concept from (a) activated sodium, and (b) fission products resulting from fuel failure. Shielding thicknesses (given in Figs. 7, 10, and 11) are shown as required to reduce radiation from sodium to levels of 100 mR/hr or less at a distance of 1 ft from the shielding surface. In case of a fuel failure, the preliminary analysis that follows shows an increase in the radiation levels by factors of 5 to 10. Further analysis may show the desirability of restricting the fueled loop to the internal-pump concept, where loop-sodium volume outside the primary-tank shielding can be minimized or eliminated.

Failure of fuel cladding would permit the release of a fraction of fission products to the flowing sodium in the loop. The severity of such an incident depends greatly on operating history, fuel type, nature and amount of fission products released, and distribution of released fission products in the loop. It is desired to estimate the increase in radiation level in the vicinity of the external components of the packaged loop immediately following the failure. The situation that has been examined is the one in which a gross cladding failure occurs after operation for an extended period of time at a packaged-loop power level of 400 kW. For this situation, the following assumptions have been made:

a. The fission products that are most likely to be released and that will contribute to the bulk of the activity are the noble gases, the halogens, and cesium. A 25% release of these isotopes is assumed.

b. The integrity of the loop is preserved, and the released fission products are uniformly dispersed throughout the entire sodium volume (approximately 8 gal) of the packaged loop.

The bulk of the radiation external to the packaged-loop shielding immediately following a failure will be due to the hard ( $>1.0$  MeV) gamma-ray emitters. Most of these radioisotopes have half-lives of less than 1 hr; hence the activity will decay very rapidly during the first day following the failure and shutdown. Further, the activity due to these short-lived isotopes is essentially independent of irradiation time.

In the equipment package with the external pump, the major components that will contain fission products are:

- a. Surge tank, approximately 2 gal
- b. Pump, 1.2 gal
- c. Cold trap, 1 gal
- d. Hot trap, 1 gal.

The radiation level resulting from fission products immediately after shutdown will decay very rapidly. A few hours after shutdown, the radiation level outside the packaged-loop shielding will primarily result from the activated sodium. Sodium-24 has a relatively long half-life (14.9 hr) and decays by emitting 2.76-MeV gamma rays. These gamma rays are very penetrating and will govern the radiation level for several days after shutdown. The radiation level outside the packaged-loop shielding is expected to decay to a level of less than 10 mR/hr about five days after shutdown.

Deliberate operation with vented or exposed fuel will result in a gradual buildup of fission products in the coolant, which can be monitored during the test. A more rigorous shielding analysis will be needed during the detailed design phase.

## 2. Shielding during Loop Handling

Reference 1 indicated that radiation problems during the removal of a completed experiment were one of the problem areas of the packaged-loop concept. The bottom of the loop coffin, when in a position to receive the experimental test section, is about  $11\frac{1}{2}$  ft above the reactor-building floor. Reference 1 concluded that it was desirable to use distance and personnel shielding for radiation protection during part of the removal operation. About  $1\frac{1}{2}$  min would be required to raise the test section through the unshielded space. Adjacent control-rod-drive mechanisms and other equipment make it impossible to place adequate shielding immediately around the withdrawal path. A study has been performed on the possibility of providing a large shielding enclosure to allow a safer removal operation.

Above the control-rod-drive scram clutches, which are about halfway from the top of the rotating plugs to the bottom of the coffin,

sufficient shielding can be placed immediately around the containment tube. Shielding pieces may be fitted around adjacent control-rod mechanisms for adequate 360° coverage, as shown in Fig. 16.

Below the scram clutches, the adjacent bellows, gripper-jaw drive motors, scram-clutch housing, etc., make it impossible to provide adequate 360° shielding coverage immediately adjacent to the withdrawal path. Because of gamma scattering, shielding only a 120° sector is not successful. Shielding for the lower section can be placed around the whole group of control-rod-drive mechanisms, the reactor-vessel cover support and elevation structure, and the gripper and holddown columns. A possible shielding configuration is shown in Fig. 16. An estimated weight for the shielding to attenuate 2- to 4-MeV gammas by a factor of  $10^5$  is 35 tons. This weight could be supported on beams that span the rotating plugs, as shown in the figure. Placing the shielding would have some effect on the EBR-II plant factor. However, this removal operation would not be expected to occur more than once or twice a year.

#### J. Loop Heat-rejection Capabilities and Temperature Distribution in Fueled Test Section

##### 1. Heat-transfer Studies: Internal-pump Concept

A preliminary study was made to investigate the heat-rejection capabilities of the packaged loop. Multitube and concentric-tube heat exchangers were considered. The multitube type consists of 12 tubes, which extend downward from the bottom of the reactor-vessel cover to the upper reactor support grid; heat is rejected to the primary coolant flowing through the reactor. In the concentric-tube type, heat is rejected from an outer annulus to the primary coolant flowing through the reactor and to the stagnant bulk sodium above the reactor vessel.

Preliminary heat-transfer calculations have been made for the concentric-tube exchanger with the internal pump. Reference 1 presented similar data for this type of exchanger with the external pump. The following standard heat-transfer equations were used for the conduction of heat through the outer wall of the annulus and for the heat lost by the loop sodium:

$$Q = UA\Delta T_1$$

and

$$Q = WC_p\Delta T_2,$$

where

$$\frac{1}{U} = \frac{r_m}{r_o h_o} + \frac{r_o - r_i}{k_m} + \frac{r_m}{r_i h_i},$$

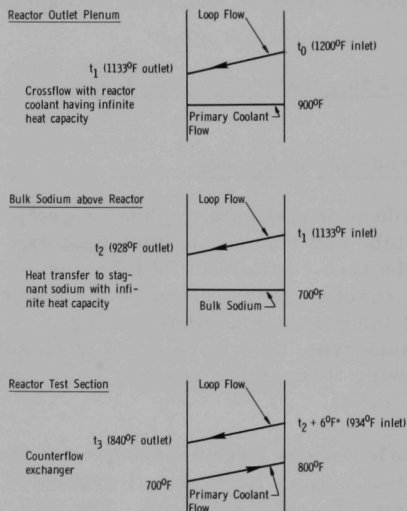
$$\Delta T_1 = \frac{\Delta T_i + \Delta T_o}{2} = \frac{(1200 - 900) + (1133 - 900)}{2} = 267^\circ\text{F},$$

and

$$\Delta T_2 = T_i - T_o = 1200 - 1133 = 67^\circ\text{F}.$$

For reactor-outlet-plenum case shown below

Since the two equations for  $Q$  represent the same quantity of heat after assuming a test-section outlet temperature, the two unknowns (heat-exchanger exit temperature and quantity of heat transferred) can be readily calculated. Temperature profiles through each of the three heat-exchanger sections are shown in Fig. 18. Temperatures in parentheses are for a test-section outlet temperature of  $1200^\circ\text{F}$  and a loop flowrate of 35 gpm.



\*There is a temperature increase of about  $6^\circ\text{F}$  through the pump.

Fig. 18. Temperature Profiles for Concentric-tube Heat Exchanger with Internal Pump

and

$$\beta = \frac{\rho_\infty - \rho}{\rho (T - T_\infty)};$$

and

$$\frac{hD_H}{k} = 0.105Gr^{1/3}Pr^m,$$

One area of uncertainty in the study is the determination of the heat-transfer film coefficient when the sodium is stagnant. Little experimental work has been done in this area. The following two equations<sup>20,21</sup> have given results that check reasonably well:

$$\frac{hD_H}{k} = \frac{0.0295Gr^{2/5}Pr^{7/15}}{(1 + 0.494Pr^{2/3})^{2/5}},$$

where

$$Gr = \rho^2 g \beta (T - T_\infty) \frac{L^3}{\mu^2}$$

where

$$m = 0.3 + \frac{0.02}{Pr^{1/3}}$$

Film coefficients of 3520 and 2240 Btu/hr-ft<sup>2</sup>-°F result from the first and second equations, respectively. The higher of these coefficients has been used because the primary bulk sodium may have some slight velocity beyond that due to natural circulation. This would be caused by the primary coolant discharge from the intermediate heat exchanger, at an elevation near the top of the reactor vessel, and by the suction of the two reactor coolant pumps.

The following equation<sup>22</sup> has been used for crossflow in the reactor outlet plenum:

$$\frac{hD_H}{k} = 0.61 \left( \frac{\phi_1}{D} \right)^{1/2} Pe^{1/2}$$

where

$$\frac{\phi_1}{D} = 2 \text{ for a single cylinder.}$$

The following general equation,<sup>23</sup> used for EBR-II heat-transfer work, has been used for all other cases:

$$\frac{hD_H}{k} = 2.3 + 0.23Pe^{1/2}.$$

Heat-exchanger lengths and heat-transfer areas are as follows:

Heat-exchanger Section	Length, ft	Area, A, ft <sup>2</sup>
1 (outlet plenum)	1.50	0.942
2 (stagnant sodium)	4.50	2.670
3 (test section)	5.42	2.920

Film and total heat-transfer coefficients, temperatures, and heat transferred resulting from the above equations are tabulated in Tables VII and VIII.

Increasing the loop coolant velocity has a significant effect on the film coefficients. However, the resistance of the 0.100-in. stainless steel wall dwarfs this increase, resulting in a very small increase in the

total heat-transfer coefficient, even though the velocity is doubled. The advantage of the higher flowrate is to decrease the temperature differential across the test section. The data in TABLE IX show that more than 535 kW can be rejected from the system with the higher core-outlet temperatures and higher flowrates.

TABLE VII. Heat-transfer Coefficients

Heat-exchanger Section <sup>a</sup>	Loop Flow-rate, gpm	Film Coefficients, Btu/hr-ft <sup>2</sup> -°F		Total Heat-transfer Coefficient, U, Btu-hr-ft <sup>2</sup> -°F	UA, Btu/hr-°F
		$h_i$ (inside)	$h_o$ (outside)		
1	35	11,750	5,280	1120	1054
2	35	12,400	3,520	1020	2720
3	35	13,800	12,100	1215	3550
1	50	13,400	5,280	1135	1068
2	50	14,100	3,520	1030	2750
3	50	15,700	12,100	1230	3600
1	70	15,090	5,280	1145	1115
2	70	16,750	3,520	1045	2790
3	70	17,800	12,100	1245	3640

<sup>a</sup> 1 = outlet plenum; 2 = stagnant sodium; and 3 = test section.

TABLE VIII. Loop Temperature Distribution and Heat Transferred

Heat-exchanger Section <sup>a</sup>	Flow-rate, gpm	Core Outlet Temp, °F	$\Delta T_1$ , °F	Q, Heat Transferred		Temp, °F	
				Btu/hr x 10 <sup>-3</sup>	kW	In	Out
1	35	1200	267	276	82	1200	1133
2	35	1200	330	898	263	1133	928
3	35	1200	137	486	142	934	840
				Total	487		
1	35	1100	179	185	55	1100	1057
2	35	1100	273	743	218	1057	889
3	35	1100	110	390	114	895	824
				Total	387		
1	35	1000	89	92.3	28	1000	978
2	35	1000	213	580	170	978	848
3	35	1000	81	287	84	854	807
				Total	282		
1	35	900	0	0	0	900	900
2	35	900	154	419	123	900	808
3	35	900	51	181	53	814	788
				Total	176		
1	50	1200	281	294	86	1200	1150
2	50	1200	398	1003	294	1150	980
3	50	1200	238	670	197	986	886
				Total	577		
1	70	1200	275	314	92	1200	1162
2	70	1200	365	1110	326	1162	1032
3	70	1200	186	866	254	1038	938
				Total	672		

<sup>a</sup> 1 = outlet plenum; 2 = stagnant sodium; 3 = test section.

If desirable, the heat rejection capacity at the lower core-outlet temperatures can be increased in the following ways:

a. Use of the multitube heat exchanger will increase the heat-transfer area within the reactor vessel. Twelve 0.375-in.-OD tubes (wall thickness of 0.060 in.) will reject 535 kW of heat when the test-section outlet temperature is 1200°F. However, the safety analysis made does not cover this type of configuration. Much of the advantage may be lost if outer pressure-containment boundaries become thicker to satisfy safety requirements.

b. The 3-ft length through the reactor-vessel cover might be used as additional heat-exchange area for the concentric-tube design. If the in-pile tube is undersized to allow the passage of about 1/3% of the reactor coolant flow, a substantial increase in heat-rejection capacity should be realized. The effect of this on the exchanger above the reactor cover would probably be small. A clearance of 0.080 in. on the radius allows a flow of 32 gpm, which will exchange an additional 100 kW when the test-section outlet temperature is 1200°F or 23 kW when the temperature is 1000°F.

The schematic flow diagram of Fig. 2 identifies heat-transfer areas that are assumed to be insulated for this study. The core section is left uninsulated for dissipating decay heat in case of a loss of loop flow. Sodium temperature is controlled by means of electric heaters and by varying the loop flowrate. For some experiments, it may be desirable to remove some insulation, between the points where flow leaves and returns to the core, to preheat the sodium entering the test section.

## 2. Heat-transfer Studies: External-pump Concept

Additional heat-transfer studies would be required in connection with the heat-exchanger development work of paragraph VI.E above. Insulation of the in-pile tube and the external loop piping from their surroundings would be studied. A very good insulation\* would be required to maintain elevated temperatures in external loop components. Results of a preliminary study for the external pump arrangement are tabulated in Table IX. Temperatures at various points in the loop are shown for seven different test conditions. Loop power varies from 200 to 400 kW with a flowrate of 35 gpm. The table assumes an electrical heater input of 20 kW and takes into account heat losses from the 13.5-in. core section and losses from all other parts of the system. It is estimated that 180 to 360 kW of heat must be rejected to the primary-tank bulk sodium and the sodium flowing through the reactor vessel. The outer annulus of the heat exchanger is insulated from the inner region by a thin gas annulus. For high-temperature experiments, it may be advantageous to allow the inlet coolant to be preheated in this area.

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\*Such as that manufactured by Union Carbide Corporation under the name of Linde Super Insulation Systems.

TABLE IX. Estimated Temperatures for Fueled  
Packaged Loop with External Pump

Test-section Temp, °F		Heat-exchanger Temp, °F		Heater-section Temp, °F		Test Subas- sembly Power, kW	Heat-exchange Capacity, kW
Inlet	Outlet	Inlet	Outlet	Inlet	Outlet		
875	1200	1200	906	901	918	400	360
775	1100	1100	903	898	915	400	365
956	1200	1200	988	983	1000	300	260
756	1000	1000	786	781	798	300	270
1034	1200	1200	1070	1065	1082	200	160
834	1000	1000	862	857	874	200	170
734	900	900	753	748	765	200	180

Heat losses will be much less with the internal pump arrangement where the loop sodium is within the primary tank. No heat-transfer work has been done for the nonfueled version of the loop.

### 3. Temperature Distributions in Fueled Test Section

Sodium, cladding, and fuel temperature distributions through the fueled region of the test section have been calculated with the HECTIC Code. Data at flowrates of 35, 50, and 70 gpm are plotted in Figs. 19-24 for both the center and outside pins and channels. All data are for a 19-pin assembly

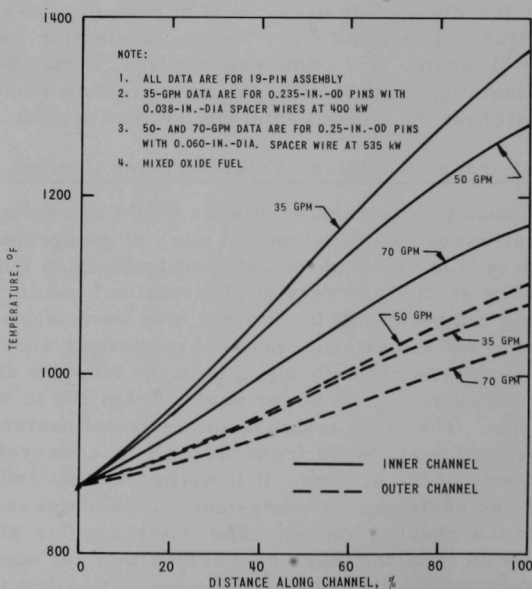


Fig. 19. Sodium-coolant Temperature Distribu-  
tion through Fueled Section of Core

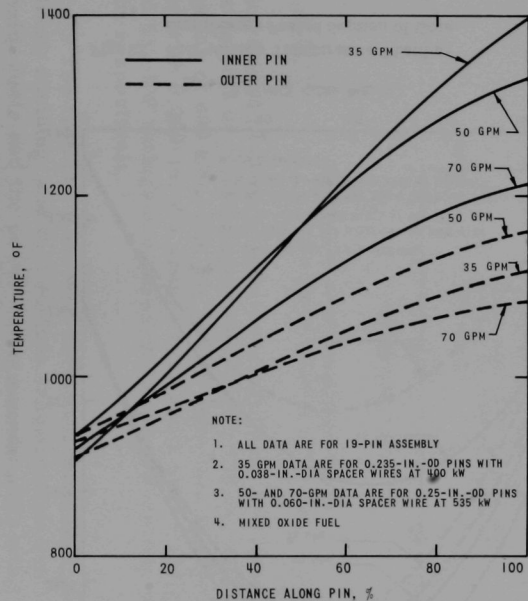


Fig. 20. Outer-cladding-surface Temperature Distribution through Fueled Section of Core

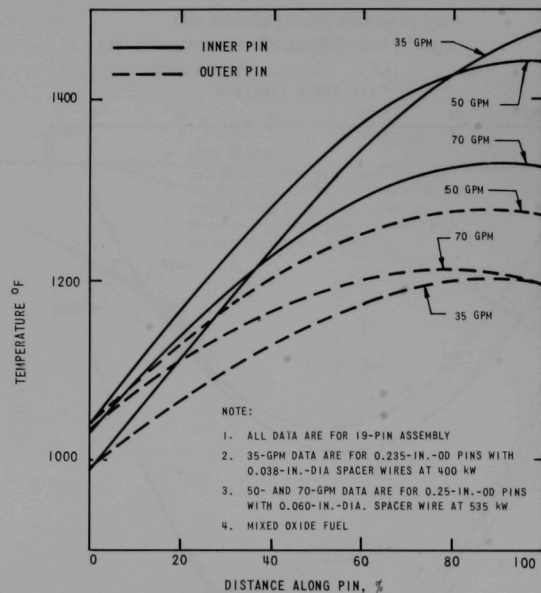


Fig. 21. Inner-cladding-surface Temperature Distribution through Fueled Section of Core

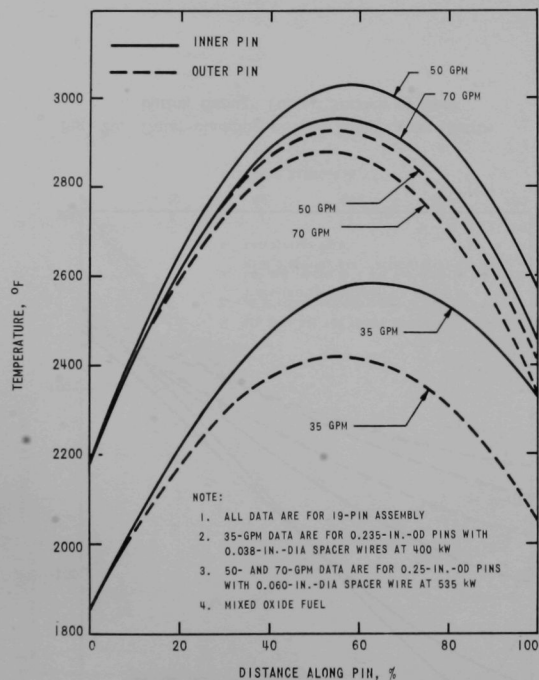


Fig. 22. Fuel-surface Temperature Distribution through Fueled Section of Core

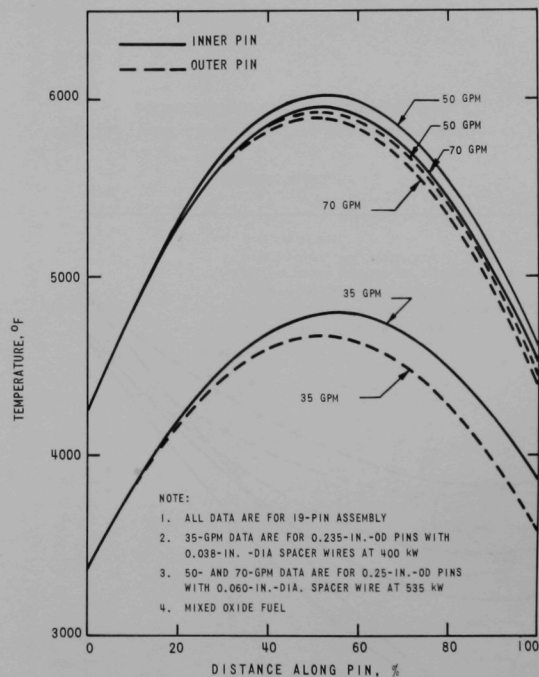


Fig. 23. Fuel-average Temperature Distribution through Fueled Section of Core

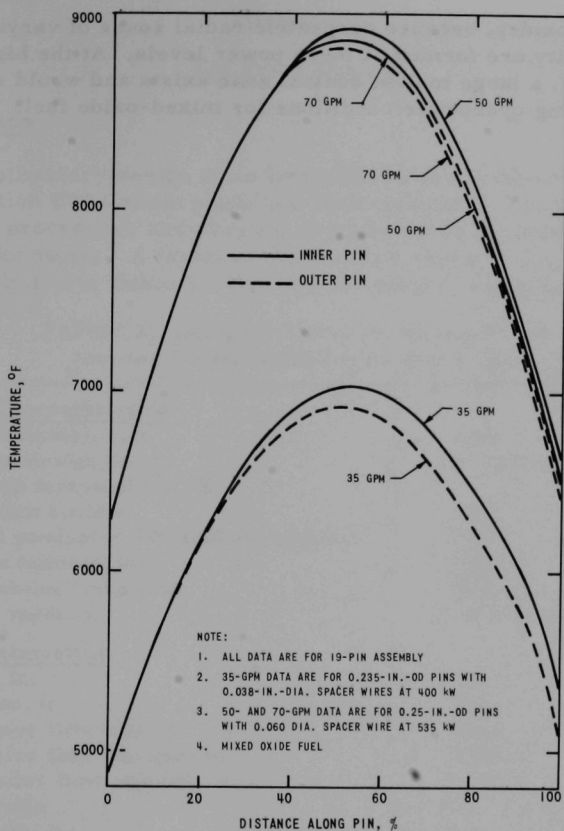


Fig. 24. Fuel-center Temperature Distribution through Fueled Section of Core

containing mixed plutonium-uranium oxide fuel. The 35-gpm data are for FFTF-size pins of 0.235-in. diameter with 0.038-in.-dia spacer wire and a power rating of 400 kW. The 50- and 70-gpm data are for pins of 0.25-in. diameter with 0.060-in.-dia spacer wire and a power rating of 535 kW. The 400- and 535-kW ratings are based on linear powers per element of 19 and 25 kW/ft, respectively.

The HECTIC Code was developed for treating metallic fuels. The sodium, cladding, and fuel surface temperatures are realistic for the mixed-oxide fuels and the power ratings used in this analysis. However, the fuel-average and -center temperatures are high and not realistic. The code allows a single conductivity value for the fuel. This is not representative

of the mixed oxides, because concentric radial zones of varying structure and conductivity are formed at these power levels. At the higher linear power ratings, a large molten central zone exists and would represent tests simulating overpower conditions for mixed-oxide fuel.



## APPENDIX

### EM Pump Design

#### 1. Introduction

A preliminary design of an internal Einstein-Szilard type of annular linear-induction EM sodium pump has been completed with the use of the basic design procedures and criteria of Refs. 24-30 to establish configuration and performance. A layout of the pump is shown in Fig. 8, and design criteria are listed in Table X. The special design problems encountered

TABLE X. Design Criteria for Reverse-flow  
Annular Linear-induction Sodium Pump

<u>Operating Characteristics</u>	
Flow (variable), gpm	5-40
Optimum design point	35 gpm at 75 psi
Maximum temperature, °F	
Sodium coolant	1200
Coil conductor (SS-clad Zr copper)	1600
Flux laminations	1400
Electrical insulation	1600
Cooling medium	700°F bulk sodium
<u>Size Characteristics</u>	
Length, in.	60
Diameter, in.	3.28
ID of inner flow tube, in.	0.625
ID of outer flow annulus, in.	1.988
OD of outer flow annulus, in.	1.790
No. of coils	66
Coil width, in.	0.5
Coil height, in.	0.375
Coil pitch, in.	0.875
No. of slots per pole per phase	2
Lamination depth, in.	
Flux return	0.542
Stator (through teeth)	0.411
<u>Power Supply</u>	
Volts (variable)	20-220
Frequency, Hz	60
Phases	3
<u>Materials</u>	
Container and flow channels	316 stainless steel
Flux laminations <sup>31-33</sup>	Hiperco 27
Coil conductors <sup>34-36</sup>	SS-clad Zr copper
Electrical insulation	Sauerisen Cement No. 8

0

were (a) a small-diameter requirement, imposed by the size of the control-rod-drive nozzle in the small rotating plug; (b) a high operating temperature, with no provision for cooling the stator windings with a low-temperature cooling medium; and (c) the need to minimize the harmonics from the fundamental traveling wave, which tends to create a braking action in the pressure annulus of the pump. These problems have been solved and pump efficiency maximized by (a) increasing the pump length, (b) using high-temperature materials that have been developed for the NASA program, (c) optimizing the arrangement of the coil, flux path, and flow annulus, (d) grading the end coils to minimize end losses, and (e) laminating the center and stator cores to keep eddy-current losses as low as possible.

The performance of an EM pump is radically changed by a number of factors, such as fluid velocity, materials, operating temperature, and stator design. Coil conductor materials greatly affect the performance of the pump as shown in Fig. 17. The highest curve is for Cube Alloy, a dispersion-hardened copper conductor containing 1% beryllia dispersed throughout. This best conductor at high temperature was not specified, because of activation of materials resulting from the  $\gamma, n$  reaction with beryllium in a gamma field. The second-best conductor, stainless steel-clad zirconium copper, was used for the pump design. The other materials selected are listed in Table X. All the materials selected are satisfactory for use at high temperatures (1200°F or greater).

The major parts of this Einstein-Szilard polyphase annular linear-induction pump are the stator and the flux-return core. The stator is on the outside of the sodium annulus and consists of the coil conductors and slotted laminations; the flux-return path or core is on the inside of the sodium annulus. The stator has a distributed winding similar to that of an ordinary three-phase motor. In this reverse-flow pump, the sodium-return path is the center tube. The pump head is developed in the outer sodium annulus.

Since completion of the pump design, higher flowrates have been considered. The characteristic curves of Fig. 17 show that for a 19-pin assembly the pump as designed is good for flows of about 40 gpm. In an experiment with fewer pins, so as to decrease the system pressure drop, this pump could provide a higher flowrate. With a longer pump, a higher flowrate and head could be achieved.

## 2. Design Procedure

Preliminary design calculations for the annular linear-induction pump are tabulated in Table XI. The procedure used in this preliminary design was taken from Ref. 30. Although the referenced procedure is for a pump with a flat, rectangular flow channel, it has been adapted for use

in designing a pump with an annular flow channel. Figure 25 is a cross section through a pump coil (Section C-C of Fig. 8) showing pump dimensions. The electrical diagram of Fig. 26 shows how the pump is wired.

TABLE XI. Preliminary Design Calculations for 3-in. Einstein-Szilard Pump

Symbol	Meaning	Magnitude
Q	Design liquid-metal flowrate	35 gpm
$v_1$	Velocity of liquid metal	227.5 in./sec
P	Pressure to be developed by pump	90 psi
$W_c$	Width of liquid-metal path (This quantity is fixed by the mean diameter of the tube channel.)	5.928 in.
$\delta_1$	Thickness of liquid metal $\delta_1 = \frac{3.85Q}{v_1 W_c}$	0.100 in.
f	Frequency of power source	60 Hz
t	Pole pitch (final value after several trials)	5.25 in.
$\rho$	Electrical resistivity <sup>27</sup> for sodium at 1200°F	$13.75 \times 10^{-6}$ $\mu\Omega/\text{in.}$
$\rho_s$	Electrical resistivity <sup>27</sup> for Type 304 stainless steel at 1200°F	$46.0 \times 10^{-6}$ $\mu\Omega/\text{in.}$
$\delta_2$	Distance from surface of stator teeth to flux-return path $\delta_2 = 0.100 + 0.020 + 0.035$	0.155 in.
N	Number of poles produced by stator winding in air-gap flux wave	11
$v_2$	Velocity of traveling wave relative to stator $v_2 = 2tf = 2(5.25)(60)$	630 in./sec
$B_1$	Peak value required of traveling flux wave in order to develop pressure P (Rudenberg equation)	24,000 lines/in. <sup>2</sup>
	$B_1 = \left[ \frac{\rho P \left\{ \left[ \frac{1.016 \times 10^{-8} (v_2 - v_1) W_c \delta_1}{\rho \delta} \right]^2 + \left[ \frac{W_c}{t} + \frac{t}{W_c} \right]^2 \right\}}{2.22 \times 10^{-16} (v_2 - v_1) N W_c \left[ \frac{W_c}{t} + \frac{t}{W_c} \right]} \right]^{\frac{1}{2}}$	
$g_a$	Number of slots per pole per phase	2
$\phi_1$	Useful air-gap flux per pole (in maxwells) $\phi_1 = 0.636 B_1 W_c t = (0.636)(24,000)(5.928)(5.25)$	$4.75 \times 10^5$
$d_1$	Depth of stator laminations $d_1 = \frac{\phi_1}{2\phi_2 W_0}$ $\phi_2 = \text{allowable magnetic flux density for Hiperco 27 cobalt steel} = 130,000 \text{ lines/in.}^2$ $W_0 = \text{mean circumference of laminations behind slots} = 9.50 \text{ in.}$	0.192 in. required 0.216 in. allowed

TABLE XI (Contd.)

Symbol	Meaning	Magnitude
$d_1$ (Contd.)	$d_1 = \frac{4.75 \times 10^5}{(2)(1.3 \times 10^3)(9.50)}$ <p>Lamination depth<sup>a</sup> for flux return path, <math>R_3 - R_2 = 0.874 - 0.332 = 0.542</math> in. (see Fig. 25)</p>	
$W_t^{(b)}$	<p>Tooth width, with a three-phase winding, two slots per pole per phase, tooth, and slot widths</p> $W_t = \frac{t}{12} = 0.437 \text{ in.}$	0.375 in. <sup>(b)</sup>
$W_s^{(b)}$	Slot width	0.500 in. <sup>(b)</sup>
$C_1$	Dimensionless quantity entering into calculation of Carter coefficient	0.607
	$C_1 = 1 - 0.636 \arctan \frac{W_s}{2\delta} + 0.636 \frac{\delta}{W_s} \log_e \left[ 1 + \left( \frac{W_s}{2\delta} \right)^2 \right]$	
$J_1$	Initial value of current density in coil conductor	5000 A/in. <sup>2</sup>
	Assume $J_1 = \sim 5000$ A/in. <sup>2</sup>	
$C_2$	Carter coefficient, used in calculating ampere-turns per pole necessary to set up magnetic field	1.288
	$C_2 = \frac{1 + \frac{W_s}{W_t}}{1 + \frac{C_1 W_s}{W_t}}$	
$n_1$	Peak ampere-turns per coil	988 or 1000 A-turns
	$n_1 = \frac{0.417 B_1 \delta_2 C_2}{g_a} = \frac{0.417(24,000)(0.155)(1.288)}{2}$	
$E_1$	Line voltage = 120 V (voltage across terminals = 208 V)	120 V
$K_{dh}$	Winding distribution factor	0.966
$K_{ph}$	Winding pitch factor	1.000
$n_2$	Number of turns per coil with windings Y connected	98 turns Use 99
	$n_2 = \frac{E_1 \times 10^8}{2.22 f \phi_b K_{dh} K_{ph} g_a}$ $= \frac{120 \times 10^8}{(2.22)(60)(4.75 \times 10^3)(0.966)(2)(1)} = 98 \text{ turns}$	
$I_1$	Coil magnetizing current (rms value)	7.2 A
	$I_1 = \frac{n_1}{1.414 n_2} = \frac{1000}{(1.414)(99)} = 7.2 \text{ A}$	

<sup>a</sup>The depth of the laminations in the flux return path has been established by setting the stator lamination area equal to the return-path lamination area.

<sup>b</sup>Although calculation shows average slot and tooth widths of 0.437 in., the limited diameter requires having a slot width of at least 0.500 in. to get sufficient ampere turns. A 0.375-in. tooth width gives sufficient area for the flux.

TABLE XI (Contd.)

Symbol	Meaning	Magnitude
$n_3$	Cross-sectional area of coil conductor for current density of 4000 or 5000 A/in. <sup>2</sup>	
	$n_{3a} = \frac{7.2}{4000} = 0.0018 \text{ in.}^2$	0.0018 in. <sup>2</sup>
	$n_{3b} = \frac{7.2}{5000} = 0.00144 \text{ in.}^2$	Use 0.00144 in. <sup>2</sup>
$d_2$	Bare conductor width assuming square conductor is used	
	$d_2 = (n_3)^{1/2} \quad 4000 \text{ A/in.}^2$	0.042 in.
	Use square wire $5000 \text{ A/in.}^2$	Use 0.038 in.
$d_3$	Overall conductor width allowing a total of 0.005 in. for insulation (ceramic)	
	$d_3 = d_2 + 0.005 \text{ for } 5200 \text{ A/in.}^2 \text{ (see } J_2, \text{ p. 94)}$	0.043 in.
$d_4$	Coil width over conductors allowing 0.020-in. thickness for side insulation	
	$d_4 = W_8 - 0.020$	0.480 in.
$n_4$	Number of conductors in coil width	
	$n_4 = \frac{d_4}{d_3} = \frac{0.480}{0.043} = 11.175 \text{ or}$	11 conductors
$n_5$	Number of conductors in slot depth	
	$n_5 = \frac{n_2}{n_4} \frac{99}{11} = 9$	9 conductors
$d_5$	Coil depth over conductor	
	$d_5 = n_5 d_3 = (9)(0.043)$	0.387 in.
$d_6$	Slot depth	
	$d_6 = d_5 + 0.031$	0.418 in.
$\delta_3$	Thickness of annulus walls (inner and outer) = 0.020 + 0.035	0.055 in.
$W_1$	Power input to sodium	
	$W_1 = 0.435PQ \left(1 + \frac{v_2}{v_1}\right) = 0.435(90)(35)(1 + 2.78)$	5175 W
$W_2$	Power loss in annulus wall	
	$W_2 = \frac{2.5 \times 10^{-17} v_2^2 N \delta_3 W_c^2 \left(\frac{W_c}{t} + \frac{t}{W_c}\right) B_1^2}{\rho_s \left[ \left( \frac{1.02 \times 10^{-8} v_2 \delta_3 W_c}{\rho_s \delta} \right)^2 + \left( \frac{W_c}{t} + \frac{t}{W_c} \right)^2 \right]}$	1095 W
$W_3$	Power loss in stator and flux-return-path iron	
	$(5 \text{ W/lb})(108.5 \text{ lb}) = 542 \text{ W}$	542 W
$W_4$	Power loss in copper in stator winding, since total coil current cannot be determined until all losses are known	
	Assume $W_4 = 2W_3$	1084 W

TABLE XI (Contd.)

Symbol	Meaning	Magnitude
$W_5$	Total power input $W_5 = W_1 + W_2 + W_3 + W_4$ $W_5 = 5175 + 1095 + 542 + 1084 = 7896$	7896 W
$I_2$	Energy component of coil current (rms value) $I_2 = \frac{W_5}{3NE_1}$ $I_2 = \frac{7896}{(3)(11)(120)}$	2.01 A
$I_3$	Total coil current (rms value) $I_3 = \sqrt{I_1^2 + I_2^2}$	7.47 A
$J_2$	Final value of current density in coil conductor $J_2 = \frac{I_3}{n_3} = \frac{7.47}{0.00144}$	5200 A/in. <sup>2</sup>
$e_1$	Efficiency of pump $e_1 = \frac{0.435 PQ}{W_5}$ $e_1 = \frac{(0.435)(90)(35)}{7896}$	17.35%

CROSS SECTION C-C OF FIG. 8

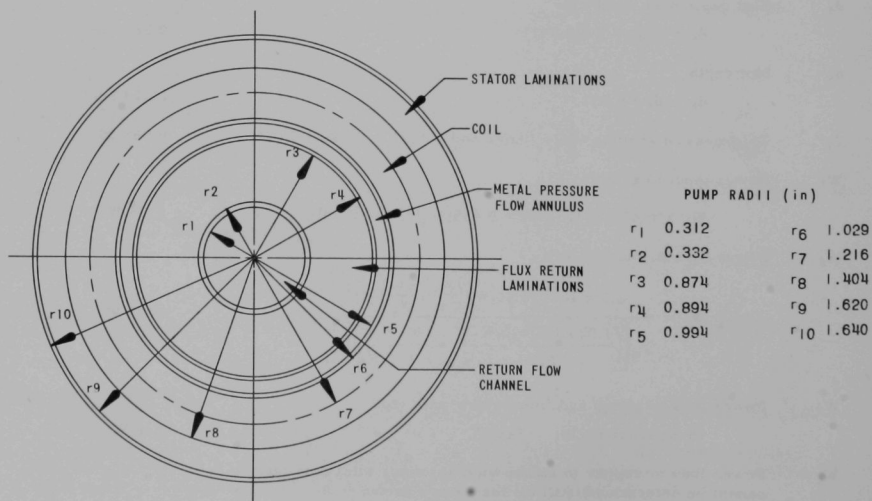


Fig. 25. Radial Dimensions of Internal EM Pump

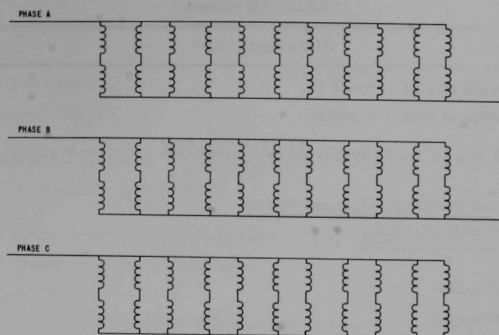


Fig. 26. Electrical Diagram of Internal EM Pump

### 3. Performance Calculations

Performance calculations for the annular linear-induction pump are tabulated in Table XII. The performance for the pump is obtained by solving the approximate equivalent circuit shown on p. 98. The procedure involves the use of a conventional alternating-current analysis and was taken from Ref. 24. The performance calculations are based on the materials identified in Table X, including Cube Alloy for the coil conductors. As indicated in Section 1 of this appendix, this material would probably not be used for the coils.

TABLE XII. Performance Calculations for 3-in. Einstein-Szilard Pump

Symbol	Meaning	Magnitude
<u>Primary-circuit Resistance</u>		
$R_1$	$R_1 = \frac{\rho_{cu} 2\pi r m c g a n_2 N}{A_{cu} a^2}$	0.245 $\Omega$
	$\rho_{cu} = 2.56 \times 10^{-6} \mu\Omega/\text{in.}$ , resistance of Cube Alloy at 1600°F	
	$r$ = mean coil radius = 1.216 in.	
	$ga$ = 2 slots per pole per phase	
	$n$ = 99 primary conductors in series per phase	
	$N$ = 11 poles	
	$A_{cu}$ = area of conductor = 0.00144 in. <sup>2</sup>	
	$a$ = 11 identical parallel paths	
	$R_1 = \frac{(2.56 \times 10^{-6})(2\pi)(1.216)(2)(99)(11)}{(0.00144)(11)^2} = 0.245$	
<u>Primary-circuit Leakage Reactance</u>		
$X_L$	$X_L = 1.26 \times 10^{-6} f n_2^2 g a \frac{N}{a^2} \cdot \frac{R_4 d_6}{3 W_s}$	0.0391 $\Omega$
	$r_6$ = inner radius of the coil = 1.029 in.	

TABLE XII (Contd.)

Symbol	Meaning	Magnitude
$X_L$ (Contd.)	$d_s$ = depth of slot = 0.418 in. $w_s$ = width of slot = 0.500 in. $X_L = (1.26 \times 10^{-6})(60)(99)^2(2) \frac{11}{121} \frac{(1.029)(0.418)}{(3)(0.500)}$	
<u>Equivalent Primary Turns per Phase</u>		
$D_h$	$D_h = \frac{N n_2 g a K_p K_d C_{t_h} C_3}{a}$ $N$ = 11 poles $n_2$ = 99 turns $ga$ = 2 slots per pole per phase $K_p$ = pitch factor = 1.000 $K_d$ = winding distribution factor = 0.966 $C_3$ = end-effect factor = $\sqrt{\frac{N-1}{N}} = \sqrt{\frac{11-1}{11}} = 0.909$ $C_{t_h}$ = tooth-effect factor $h$ = order of space harmonics of the magnetic field = 1, 5, 7, 11, 13, and 17 $D_h = \left[ \frac{(11)(99)(2)(1)(0.966)(0.909)}{11} \right] C_{t_h} = 174 C_{t_h}$ $C_{t_h} = 1 + \frac{C_2 - 1}{\frac{36}{h^2} - 1}$ $C_2$ = Carter coefficient $C_{t_1} = 1 + \frac{0.288}{\frac{36}{1} - 1} = 1.0082$ $C_{t_5} = 1 + \frac{0.288}{\frac{36}{25} - 1} = 1.665$ $C_{t_7} = 1 + \frac{0.288}{\frac{36}{49} - 1} = -0.095$ $C_{t_{11}} = 1 + \frac{0.288}{\frac{36}{121} - 1} = -0.590$ $C_{t_{13}} = 1 + \frac{0.288}{\frac{36}{169} - 1} = -0.633$ $C_{t_{17}} = 1 + \frac{0.288}{\frac{36}{289} - 1} = -0.670$ $D_1 = (1.0082)(174)$ $D_5 = (1.665)(174)$ $D_7 = (-0.095)(174)$ $D_{11} = (-0.590)(174)$	175.5 290 -16.5 -102.5

TABLE XII (Contd.)

Symbol	Meaning	Magnitude
$D_h$ (Contd.)	$D_{13} = (-0.633)(174)$ $D_{15} = (-0.670)(174)$	-110.0 -116.5
<u>Magnetizing Reactance</u>		
$X_m$	$X_m = 22.9 \times 10^{-6} \frac{rL}{N^2 C_4 \delta_2} D_1^2$ $r_s = \text{mean radius of sodium annulus} = 0.944 \text{ in.}$ $L = \text{length of pump} = 57.75 \text{ in.}$ $N = 11 \text{ poles}$ $C_4 = \text{Carter coefficient} + 10\% \text{ for iron loss} = C_2 + 0.10C_2 = 1.42$ $\delta_2 = \text{gap distance} = 0.155 \text{ in.}$ $D_1 = 175.5$ $X_m = (22.9 \times 10^{-6}) \frac{(0.944)(57.75)}{(121)(1.42)(0.155)} (175.5)^2 = 1.445$	1.445 $\Omega$
E	Assumed Line Voltage	120-V three-phase ac
<u>Nonfundamental Reactance</u>		
$X_a$	$X_a = X_m \sum_{h=5}^{h=17} \frac{(D_h/D_1)^2}{h^2}$ $\left(\frac{D_5}{D_1}\right)^2 = \left(\frac{290}{175.5}\right)^2 = 2.725$ $\left(\frac{D_7}{D_1}\right)^2 = \left(\frac{-16.5}{175.5}\right)^2 = 0.0089$ $\left(\frac{D_{11}}{D_1}\right)^2 = \left(\frac{-102.5}{175.5}\right)^2 = 0.341$ $\left(\frac{D_{13}}{D_1}\right)^2 = \left(\frac{-110}{175.5}\right)^2 = 0.392$ $\left(\frac{D_{17}}{D_1}\right)^2 = \left(\frac{-116.5}{175.5}\right)^2 = 0.442$ $X_a = 1.445 \left( \frac{2.725}{25} + \frac{0.0089}{49} + \frac{0.341}{121} + \frac{0.392}{169} + \frac{0.442}{289} \right)$ $= 1.445(0.109 + 0.00018 + 0.0028 + 0.0023 + 0.0015)$ $= 0.167.$	0.167 $\Omega$
<u>Liquid Resistance</u> $h = 1$		
$R_2$	$R_2 = 18.86 \frac{\rho r_s}{L \delta_1} D_1^2$ $\rho = \text{resistance of sodium at } 1200^\circ\text{F, } 13.75 \times 10^{-6} \mu\Omega\text{-in.}$ $r = 0.944 \text{ in. mean radius}$ $L = 57.75 \text{ in. length of pump}$ $\delta_1 = 0.100 \text{ metal gap}$ $R_2 = (18.86) \frac{(13.75 \times 10^{-6})(0.944)}{(57.75)(0.100)} (175.5)^2 = 1.31$	1.31 $\Omega$

TABLE XII (Contd.)

Symbol	Meaning	Magnitude
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Shell Resistance  $h = 1$ 

$$R_s = 18.86 \frac{\rho_s r_s}{L_1} D_1^2 \quad 7.95 \, \Omega$$

$\rho_s$  = resistance of Type 304 stainless steel at 1200°F  
 $= 46.0 \times 10^{-6} \, \mu\Omega\text{-in.}$

$r$  = mean radius = 0.944 in.

$L$  = length of pump = 57.75 in.

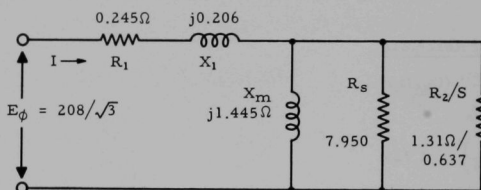
$\delta$  = shell thickness = 0.55 in.

$$R_s = 18.86 \frac{(46 \times 10^{-6})(0.944)}{(57.75)(0.055)} (175.5)^2 = 7.95$$

Equivalent Series Reactance

$$X_1 = X_L + X_a \quad 0.206 \, \Omega$$

$$X_1 = 0.0391 + 0.167 = 0.206$$

Approximate Equivalent Circuit (for one phase) $Q_p$  Synchronous Flow

Assume

Line voltage,  $E = 120 \, \text{V}$

Design liquid-metal flowrate,  $Q = 35 \, \text{gpm}$

$$Q_p = 195 t \delta_1 r_s \quad 96.6 \, \text{gpm}$$

$t$  = pole pitch = 5.25 in.

$\delta_1$  = liquid-metal thickness = 0.100 in.

$r_s$  = mean radius of sodium annulus = 0.944 in.

$$Q_p = (195)(5.25)(0.100)(0.944) = 96.6$$

Slip

$$s = \frac{Q_p - Q}{Q_p} = \frac{96.6 - 35}{96.6} \quad 0.6375$$

Air-gap Impedance

$$Z_g = \frac{jR_2 R_s X_m}{R_2 R_s + jX_m(R_2 + R_s S_1)} \quad 1.083 / 48.5^\circ$$

$$Z_g = \frac{(1.31)(7.95)(1.445) / 90^\circ}{(1.31)(7.95) + j1.445[1.31 + (7.95)(0.6375)]}$$

$$Z_g = \frac{15.05 / 90^\circ}{10.4 + j9.22} = \frac{15.05 / 90^\circ}{13.9 / 41.5^\circ}$$

$$Z_g = 0.718 + j0.810 = 1.083 / 48.5^\circ$$

TABLE XII (Contd.)

Symbol	Meaning	Magnitude
$Z_\phi$	<u>Total Impedance per Phase</u> $Z_\phi = R_1 + jX_1 + Z_g$ $Z_\phi = 0.245 + j0.206 + 0.718 + j0.810$ $Z_\phi = 0.963 + j1.016 = 1.400 \angle 46.5^\circ$	1.400 $\Omega / 46.5^\circ$
	<u>Phase Current</u>	49.49 A
$I_a$	$I_a = \frac{E}{(\sqrt{3})(Z_\phi)} = \frac{120}{(1.732)(1.400)} = 49.49 \text{ A}$	
	<u>Input Power Factor</u>	0.6884
P.F.	P.F. = $\cos 46.5^\circ = 0.6884$	
	<u>Total Input Power</u>	
$W_5$	$W_5 = 3I_a^2 R_\phi = 3(49.49)^2(0.963) = 7076$ $R_\phi = \text{resistance component of total impedance per phase}$	7076 W or 7.076 kW
	<u>Developed Pressure Calculations</u> for $h = 1$	
$P_1$	$P_1 = \frac{6.9Z_g^2 S I_a^2}{R_2 Q_p} = \frac{(6.9)(1.083)^2(0.0637)(49.5)^2}{(1.31)(96.6)}$ $P_1 = 99.7 \text{ psi}$	99.7 psi
	<u>Pressure Drop due to Harmonics</u>	
$P_h$	$P_h \cong \frac{(-m_h)6.9X_m^2 S_h I_a^2}{R_2 Q_p h^3}$ $m_h = +1 \text{ for } h = 5, 11, 17$ $= -1 \text{ for } h = 1, 7, 13, 19$ $X_m = 1.445 \Omega$ $I_a = 49.49 \text{ A}$ $R_2 = 1.31 \Omega$ $Q_p = 96.6 \text{ gpm}$ $P_h = 279(-m_h) \frac{S_h}{h^3}$ $S_h = \frac{Q_p + \frac{h}{m_h} Q}{Q_p}$ $S_5 = \frac{96.6 + 5(35)}{96.6} = 2.81$ $S_7 = \frac{96.6 - 7(35)}{96.6} = -1.53$ $S_{11} = \frac{96.6 + 11(35)}{96.6} = 4.98$ $S_{13} = \frac{96.6 - 13(35)}{96.6} = -3.71$	

<sup>a</sup>This final value of input power differs from the preliminary value given on page 94.

TABLE XII (Contd.)

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